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SYSTEMATIC EVALUATION PROGRAM

STATUS REPORT AND INITIAL EVALUATION

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EXECUTIVE SUMMARY

Background

The MHB Ongoing Systematic Evaluation Program (SEP) Assessment Study was initiated by the Swedish Nuclear Power Inspectorate (SKI) in 1980. In January, 1981, the Study was extended due to potentially significant changes in SEP caused by Congressional action. The extended SEP Assessment Study elements are contained in Appendix F to this report.

Scope of Report

Three aspects of the SEP Study have not developed sufficiently to review. First, the impact of Congressional legislation on SEP and the NRC's safety review of nuclear power plants has not become evident at this time. Second, the NRC's planned extension of SEP to review all operating reactors (known as SEP Phase III) is not yet underway. Finally, since it is not clear what effect SEP will have on nuclear plants (i.e., how many and what type of final modifications will be required) an analysis of the ultimate value of SEP would be premature.

The above three aspects of SEP, which would be included in a final assessment report on SEP, cannot be included in this report. However, since the NRC is beginning to publish the first plant review reports under the Systematic Evaluation Program, MHB feels that a status report at this time is important and worthwhile.

Therefore, this MHB report is intended as a status report and initial evaluation of the Systematic Evaluation Program. The report discusses the methodology and results of SEP, with particular emphasis on the first two SEP plant reviews--the Palisades and R. E. Ginna nuclear power plants. The comments of cognizant persons in the NRC and the ACRS, as well as private consultants, are included herein. The report identifies the remaining outstanding issues and concerns for the Systematic Evaluation Program.

Major Findings

MHB's major findings, as discussed more fully in the sections of this report, are as follows:

- The SEP plant review methodology was acceptable to the NRC Commissioners, the ACRS, and the NRC Staff's consultants who evaluated the first two SEP plant reviews.
- A concern raised by all who commented on SEP was the absence of Three Mile Island Action Plan Items and Unresolved Safety Issues from current SEP reviews.
- The SEP reviews of the Palisades and R. E. Ginna plants concluded that the two plant designs were adequate with respect to a majority of safety topics.
- Several topics remain unresolved in both the Palisades and R. E. Ginna SEP reviews. In the case of the Ginna plant, several related topics have been grouped together in a major structural reevaluation study.
- In general, due to the number of unresolved and excluded topics, SEP has not at this time produced a plant safety evaluation which can be considered complete and integrated.

SYSTEMATIC EVALUATION PROGRAM

STATUS REPORT AND INITIAL EVALUATION

DRAFT REPORT

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SECTION 1

THE SYSTEMATIC EVALUATION PROGRAM

1.1: SYSTEMATIC EVALUATION PROGRAM HISTORY AND OBJECTIVES

The Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission (NRC) in 1977 to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety.[1/] The review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.[2/]

The original SEP objectives were:

(1) The program should establish documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant safety issues and should provide a rationale for acceptable departures from these criteria.

(2) The program should provide the capability to make integrated and balanced decisions with respect to any required backfitting.

(3) The program should be structured for early identification and resolution of any significant deficiencies.

(4) The program should assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

(5) The program should efficiently use available resources and minimize requirements for additional resources by NRC or industry. 3/

The program objectives were later interpreted to ensure that the SEP also provides safety assessments adequate for conversion of provisional operating licenses (POL's) to full-term operating licenses (FTOL's).

1.2: SYSTEMATIC EVALUATION PROGRAM, PHASE I

The SEP review compares the as-built plant design with current review criteria from a list of different areas defined as "topics". Phase I of the SEP comsisted of the formation and refinement of the SEP Topic List. Originally, the List contained over 800 topics.[4/] The List was further reviewed and duplications of topics were removed, leaving 137 topics.[5/] Topics which were being reviewed as Unresolved Safety Issues (USI) or Three Mile Island (TMI) Action Plan Tasks were then eliminated from the Topic List.[6/] The Final SEP General Topic List contained 113 review areas.* For any given plant, some topics may not be applicable. Thus, the Palisades Nuclear Plant review, for example, consisted of ninety topics.[7/]

1.3: SYSTEMATIC EVALUATION PROGRAM, PHASE II

Phase II of the SEP consists of applying the Topic List to the review of a group of nuclear power plants. Eleven plants

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* The 137 SEP topics are included herein as Appendix A. Asterisks mark the 24 topics which are being reviewed as USI's or TMI Tasks.

were chosen for review. They include five of the oldest nuclear reactor plants and seven plants under NRC revi4w for the conversion of POL's to FTOL's. Table 1.3-1 shows the plants selected for the original Phase II program.

TABLE 1.3-1

SYSTEMATIC EVALUATION PROGRAM, PHASE II PLANTS

Plant Name	Type[8/]	Type of License[9/]	Design Electric Rating (MWe-net)[10/]	Date of Commercial Operation [11/]
Yankee Rowe	PWR	FTOL	175	July 1, 1961
Haddam Neck	PWR	FTOL	582	Jan. 1, 1968
Millstone 1	BWR	POL	660	March 1, 1971
Oyster Creek	BWR	POL	650	Dec. 1, 1969
Ginna	PWR	POL	470	June 1, 1970
Lacrosse	BWR	POL	50	Nov. 1, 1969
Big Rock Point	BWR	FTOL	72	March 29, 1963
Palisades	PWR	POL	805	Dec. 31, 1971
Dresden 1	BWR	FTOL	200	July 4, 1960
Dresden 2	BWR	POL	794	June 9, 1970
San Onofre	PWR	POL	436	Jan. 1, 1968

The SEP review of Dresden 1 has been deferred because the plant is undergoing an extensive modification and is not scheduled for restart before June, 1986.[12/]

Phase II reviews are proceeding at this time. Section 2 discusses the review method being used by the NRC. The NRC plans to extend the SEP review process to include all licensed nuclear power plants. This program will be called Phase III and will begin after Phase II is complete. Phase III is expected to require seven years for completion.[13/] Since the Phase III reviews are not yet underway, Phase III will not be a part of this report.

SECTION 1

LIST OF REFERENCES

1/ NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant, U.S. Nuclear Regulatory Commission, Washington, D.C., April, 1982, p. 1-2.

2/ Ibid 1 p. 1-2.

3/ Ibid 1, p. 1-2.

4/ Transcript of April 15, 1982 Advisory Committee on Reactor Safeguards Subcommittee on the Systematic Evaluation Program meeting, U.S. Nuclear Regulatory Commission, ACRS, Washington, D.C., p. 12.

5/ Ibid 4, p. 13.

6/ Ibid 1, Appendix B.

7/ Ibid 1, pp. 3-1 to 3-5.

8/ NUREG-0020, Licensed Operating Reactors Status Summary Report, U.S. Nuclear Regulatory Commission, Washington, D.C., May, 1982, Section 2. (For information on Dresden 1 the January, 1977 edition of NUREG-0020 was used.)

9/ Ibid 1, p. 1-2 to 1-3.

10/ Ibid 8.

11/ Ibid 8.

12/ Ibid 1, p. 1-3.

13/ Transcript of October 22, 1981 Commission meeting, U.S. Nuclear Regulatory Commission, Washington, D.C., p. 74.

SECTION 2

THE SYSTEMATIC EVALUATION PROGRAM REVIEW PROCESS

2.1: INTRODUCTION

The SEP Review Process consists of three elements. The first is a review of operating experience, which will be discussed in Section 2.2. The second, topic review, is the subject of Section 2.3. Integrated assessment, the third review element, will be discussed in Section 2.4. Examples of the above elements will be drawn from the two SEP reviews which are now complete. The two nuclear power plants which have been reviewed are the Palisades Nuclear Plant and the R.E. Ginna Nuclear Plant.

2.2: OPERATING EXPERIENCE REVIEW

Both the Palisades Nuclear Plant and R.E. Ginna Nuclear Plant reviews contain a summary of operating experience. The Nuclear Regulatory Commission (NRC) hired Oak Ridge National Laboratory (ORNL) to perform the operating experience reviews.[1/][2/]

ORNL's reviews focused upon forced shutdowns and power reductions, licensee event reports (LER), and enforcement actions. Table 2.2-1 summarizes the operating experience review for the Palisades and R.E. Ginna nuclear power plants. By reviewing the causes of forced shutdowns and reductions, reportable events,

TABLE 2.2-1

SUMMARY OF OPERATING HISTORY

PALISADES AND GINNA NUCLEAR PLANTS

	Palisades	Ginna
Type of License	POL	POL
Date Issued	3/71	9/69
Forced Shutdowns*	129	121
Design Basis Event Initiators	52	24
Forced Reductions*	22	45
Number of Licensee Event Reports Reviewed	340	183
% Involving Human Errors	46%	56%
Enforcement Actions**		
Civil Penalties	3	0
Commission Orders	2	0
Immediate Action Letters	3	0

Source: NUREG-0820 Section 1.4, NUREG-0821 Section 1.4, U.S. Nuclear Regulatory Commission

* For Palisades: From March 1971 to January 1980; For Ginna: From September 1969 to April 1982.

** For Palisades: From November 1979 to January 1982; For Ginna: During period of Systematic Assessment of Licensee Performance (SALP).

and enforcement actions, the NRC hopes to understand the major problems encountered and desirable features in each nuclear plant's design as demonstrated in its operating history. The plant's operating history is then factored into the overall SEP review. 3/

2.3: TOPIC REVIEW

In the topic review step, each SEP topic which applies to the nuclear power plant is reviewed to determine whether the corresponding plant design is consistent with current licensing criteria such as regulations, guides, and Standard Review Plan criteria, or the equivalent of such criteria. Safety evaluation reports (SERs) for all topics are issued to document the comparison with current licensing criteria and to identify potential areas for backfitting.

SEP has evaluated topics by two methods:

(1) The NRC staff reviews and formally issues an SER to the licensee. This SER is termed a draft because it is only one input element to the evaluation. The purpose of the draft SER is to verify the factual accuracy of the described facility and to allow the licensee to identify possible alternate approaches to meeting the current licensing criteria. After a review of the licensee's comments on the draft SER, factual changes are incorporated as needed, proposed alternatives are reviewed, and the SER is issued in final form.

(2) The licensee submits a safety analysis report and the staff issues a final SER based on a review of this submittal. 4/

SEP topic reviews prior to 1981 were characterized by the primary use of the first method. As a result of a proposal submitted by the owners of SEP Phase II plants, the NRC in a January 14, 1981 letter adopted a redirected SEP review plan relying mainly upon the second method above for topic review.5/ The second method allows the NRC to review topics and expend less Staff time, since the licensee performs a higher proportion of the analytical work.

NUREG-0485, the monthly status report on the SEP, describes the topic review process in finer detail. Seven milestones of topic review are listed in NUREG-0485. They are:

(1) Questions Received From Review Branch

This status is used for internal NRC tracking of topic review progress. The status indicates that all information necessary to perform a topic evaluation is not available and that questions have been developed and sent to the SEP branch.

(2) Questions Sent to Utility

The questions developed in the previous status have been formally transmitted to the licensee as a request for additional information.

(3) Question Response Received by NRC

The licensee has formally responded to the information request issued by NRC regarding the topic.

(4a) <u>Draft SE Received from Review Branch</u>

This status is used for internal NRC tracking of topic review progress. The status indicates that a draft topic safety evaluation is under management review.

(4b) <u>SAR Received from Licensee</u>

Licensee has submitted a Safety Analysis Report (SAR) in accordance with SEP redirection. (See Eisenhut letter to SEP owners of January 14, 1981). The SAR is under review and a Final NRC safety evaluation (status 7) is being prepared. Draft SE will not be issued for topics with licensee SAR submittal.

(5) <u>Draft SE Sent to Utility</u>

A draft topic safety evaluation has been formally transmitted to the licensee for their review and certification of the accuracy of the described facility. Licensees may comment upon NRC conclusions or propose alternatives to NRC positions.

(6) <u>SE Response Received by NRC</u>

The draft SE has been reviewed and licensee comments if any have been formally transmitted to the NRC.

(7) <u>Topic Complete</u>

The NRC topic evaluation has been completed and a final safety evaluation issued. A topic is completed in several ways: (1) The licensee has responded that the NRC draft safety evaluation (i.e. status 5) is correct, or (2) the licensees comments on the NRC draft safety evaluation (i.e. status 6) have been reviewed and if appropriate incorporated into the final safety evaluation, or (3) the licensee has not responded in the time requested, thus concurring that the draft safety evaluation is correct. 6/

In terms of the two topic review methods discussed previously, method (1) involves milestones 1, 2, 3, 4a, 5, 6 and 7 above. Method (2) involves milestones 1, 2, 3, 4b, and 7.

After completion of the topic evaluation, the disposition of each topic is grouped according to one of the following results:

(1) The plant is consistent with current licensing criteria and the topic review is considered complete. If the plant does not meet current licensing criteria, but the present design is equivalent to current criteria, the topic is also considered complete. A justification for this conclusion is provided in the topic SER.

(2) The plant is not consistent with current licensing criteria, but the licensee has implemented or proposed design or procedural changes that the staff finds acceptable.

(3) The plant is not consistent with current licensing criteria, and the differences from these criteria are to be evaluated as potential candidates for backfitting. If the staff determines the difference is of immediate safety significance, action is taken to promptly resolve the issue. No issues at Palisades required that prompt action be taken. If the difference is not of immediate safety significance, the resolution is deferred to the integrated plant safety assessment to obtain maximum benefit from co-ordinated and integrated backfit decisions. 7/

Table 2.3-1 shows, for the Palisades and Ginna plants, a summary of the topics review results.

TABLE 2.3-1

RESULTS OF TOPIC REVIEWS FOR PALISADES AND GINNA*

	Palisades	Ginna
Topics Reviewed	90	92
Acceptable	57	58
Modification During Review	2	7
Evaluated For Backfitting	31	27

2.4: INTEGRATED ASSESSMENT

The "integrated assessment" is the process by which the NRC decides how to correct differences between current review criteria and as-built conditions in a given nuclear power plant. Differences are evaluated as to their safety significance, and then the NRC and the licensee discuss possible corrective actions.

The safety evaluation of topics for which differences exist between plant conditions and current review criteria, consists of engineering judgement and limited probabilistic

*Source: NUREG-0820 Table 2.1, NUREG-0821 Table 2.1, U.S. Nuclear Regulatory Commission

risk assessment (PRA). A team of NRC reviewers evaluate the magnitude of differences based on judgements such as how serious is the effect on a system, how important is the affected system to safety, and how likely is the effect to occur. The team represents NRC programs including Inspection and Enforcement, Reliability and Risk Assessment, Licensing, and SEP. 8/

The team's judgement is supplemented by a PRA study. For instance, in the case of the Palisades SEP review a study was done by Sandia Laboratories and Science Applications Inc. (SAI). 9/ The PRA study investigated the importance to risk of thirteen issues from the Palisades Nuclear Plant integrated assessment list of thirty-one issues. 10/ Since no full PRA of Palisades had ever been done, the Sandia/SAI PRA used a limited review method to categorize issues by their importance to risk. The Sandia/SAI Study investigated the effect of an issue on the corresponding system. The importance of the issue to the system was evaluated as Low, Medium, or High. Then plant systems were similarly evaluated in their importance to overall plant risk. Table 2.4-1 shows how this was done. Finally, the two evaluations were combined to decide the importance of each issue to overall risk, as shown in Figure 2.4-1. The Sandia/SAI Study is contained in Appendix D to NUREG-0820, the SEP review of the Palisades Nuclear Plant.

TABLE 2.4-1

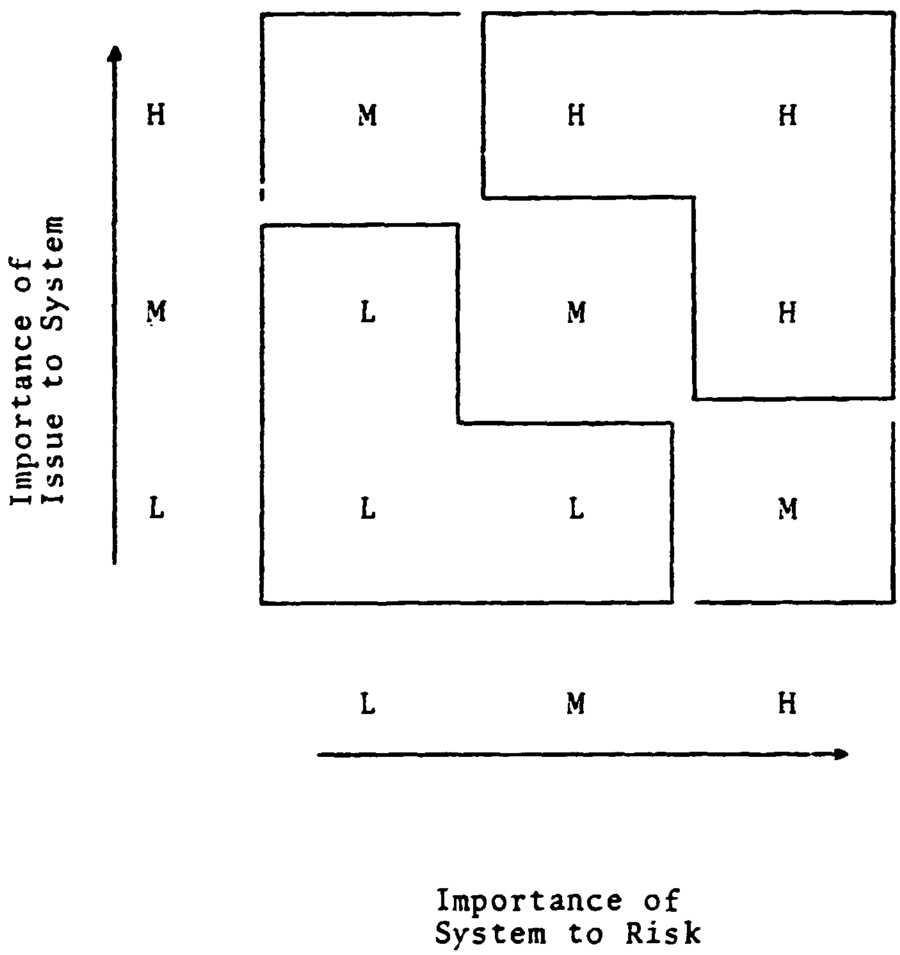
IMPORTANCE OF SYSTEMS TO RISK

	System	Relative Contribution To Risk
	Auxiliary Feedwater System	1.0
	Main Feedwater System	4×10^{-1}
	Offsite Power	4×10^{-1}
	Recovery of Main Feedwater	3×10^{-1}
H	Reactor Protection System	2×10^{-1}
	Diesel Generators	8×10^{-2}
	Batteries	5×10^{-2}
	Service Water System	5×10^{-2}
	Relief Valve (Sticks Open)	5×10^{-2}
	Transients (Other Than LOP, MFW)	4×10^{-2}
	Recovery of Stuck Open Relief Valve	4×10^{-2}
	Recovery of Offsite Power	2×10^{-2}
M	Component Cooling Water System	1×10^{-2}
	Recovery of Diesel Generators	1×10^{-2}
	High Pressure Injection System	9×10^{-3}
	Room Coolers	6×10^{-3}
	Small LOCA, S_2	5×10^{-3}
	Sump Valves	3×10^{-3}
L	Recirculation Actuation System	2×10^{-3}
	Containment Leakage	1×10^{-3}
	Safety Injection Actuation System	8×10^{-5}

Source: NUREG-0820, Appendix D, Table 5,
U.S. Nuclear Regulatory Commission

FIGURE 2.4-1

ISSUE IMPORTANCE-TO-RISK MATRIX



Source: NUREG-0820, Appendix D, Figure 4,
U. S. Nuclear Regulatory Commission

When the integrated assessment team has completed its evaluation of an issue's importance, the NRC then works with the licensee to determine an appropriate corrective action. If the issue is considered to be of low importance, the issue may be corrected by a change in procedures, technical specifications, or maintenance. 11/ The constructed equipment may be accepted as is, or the function may be performed using non-safety equipment. 12/

Issues having higher importance may require "backfitting", or physical changes necessary to meet review criteria retroactively.

The NRC's SEP personnel have worked with licensee representatives to seek cost-effective resolutions to the issues identified by SEP as high importance to safety.13/ The decision, for a given resolution made by the NRC, marks the completion of the SEP review process for each issue.

SECTION 2

LIST OF REFERENCES

1/ NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant, U.S. Nuclear Regulatory Commission, Washington, D.C., April, 1982, p. 1-5.

2/ NUREG-0821, Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant, U.S. Nuclear Regulatory Commission, Washington, D.C., May, 1982, p. 1-4.

3/ Ibid 2, p. 1-4.

4/ Ibid 1, p. 2-2.

5/ NRC letter to SEP licensees, January 14, 1981, U.S. Nuclear Regulatory Commission, Washington, D.C.

6/ NUREG-0485, Systematic Evaluation Program, Status Summary Report, U.S. Nuclear Regulatory Commission, Washington, D.C., May 1982, p. 1-5.

7/ Ibid 1, pp. 2-2 to 2-4.

8/ Transcript of April 15, 1982 Advisory Committee on Reactor Safeguards Subcommittee on the Systematic Evaluation Program meeting, U.S. Nuclear Regulatory Commission, ACRS, Washington, D.C., p. 30.

9/ Ibid 1, Appendix D.

10/ Ibid 1, p. D-12.

11/ Ibid 7, p. 30.

12/ Ibid 7, p. 30.

13/ Ibid 7, p. 31.

SECTION 3

PRODUCTS AND RESULTS OF THE SYSTEMATIC EVALUATION PROGRAM

3.1: INTRODUCTION

Section 3 will discuss the products and results of SEP reviews. There are three tangible products which result from a SEP review: issues evaluated as acceptable, required modifications, and documentation of the review itself. Section 3.2, 3.3, and 3.4, respectively, discuss these types of products. Section 3.5 analyzes the specific results of SEP reviews of the Palisades Nuclear Plant and the R. E. Ginna Nuclear Plant.

3.2: ISSUES EVALUATED AS ACCEPTABLE

Many plants in the United States were licensed before the existence of the NRC's Standard Review Plan. The present regulations and licensing criteria are considerably more detailed and specific than those in existence in 1970 when, for example, the Palisades plant was reviewed. The legitimate question arises as to the risks of the older licensed nuclear power plants compared to those reviewed under the Standard Review Plan.

Each SEP review distinguishes topics for which a given plant meets current review criteria or equivalent criteria from topics for which there are differences between the review criteria and the as-built plant. Therefore, one product

of the SEP review is the identification of topics for which the difference in the method of review did not affect a given plant's relative safety. The lists of acceptable topics for the Palisades and R. E. Ginna plants are included herein as Appendix B.

3.3: REQUIRED MODIFICATIONS

The NRC in the SEP review process decides what modifications are necessary to resolve differences between the as-built plant and the plant design necessary to meet current review criteria. Where modifications are required as a result of the SEP review, they are of two types: procedural and physical.

Procedural modifications include changes in technical specifications, operating procedures, surveillance and testing, or maintenance. Modifications to procedures usually resolve issues of low or moderate safety significance. Physical modifications include any changes in the as-built configuration of equipment or in the type of equipment used. Because modifications may range from minor to very significant and expensive, cost-benefit analyses are used to choose the best alternative from several proposals.

Theoretically, when the licensee has completed all modifications required by the NRC, the nuclear plant will be comparable to more recent plants in all aspects having safety significance. The modifications required for the Palisades and R. E. Ginna plants are included herein as Appendix C.

3.4: DOCUMENTATION OF SEP REVIEW

A third product of the SEP review process is the review documentation. The documentation consists of separate packets for all of the SEP topics reviewed at a given plant. Each packet contains a record of the individual topic review, integrated assessment (if any), and modification implemented (if any).

All documents, both technical and correspondence, which relate to each topic are thus located together, representing a complete record of the analysis and decision-making which led to the final result. The SEP documentation is a very important product because it provides, for any future reviewer, the basis by which the NRC considers a given plant to meet current review criteria in areas of safety significance.

3.5: PRESENT RESULTS

Section 4 of each SEP review report contains a topic-by-topic discussion of the issues for which a nuclear plant does not meet current review criteria or the equivalent. The discussion includes a statement of the difference between current criteria and the plant construction or design, an evaluation of the importance of the difference in terms of plant safety, and a statement of what the NRC will require of the licensee to resolve the difference.

In the SEP reports on the Palisades and R. E. Ginna plants, resolutions ranged from no action (when the safety significance of the difference was low), to physical modifications or in-depth studies. The differences between plant design and construction and current licensing criteria, as well as the most significant NRC requirements, are summarized below for the two plants.

3.5.1: SUMMARY OF RESULTS - PALISADES

The SEP review found that the effects of external phenomena were not considered as thoroughly during the licensing of Palisades as they would be today. The effects of flooding on flood control structures and on safety equipment, and the effects of earthquakes on piping and electrical components, are issues for which both further studies and modifications are being required of the licensee.[1/]

The review found that safety systems may not be sufficiently protected to assure their function during accident sequences such as loss of onsite power, loss of coolant accidents (LOCA), pipe breaks, and fires. Possible failures of service water for safety equipment, or ventilation for rooms housing electrical equipment, could result in vital safety equipment being subjected to intolerable environments.[2/] The NRC is requiring that either fire protection or redundant instrumentation be provided for the reactivity control and

reactor coolant instrumentation at Palisades.[3/] Further studies are being required on the subject of protection of safety equipment during LOCA's, pipe break accidents, and loss of power events.[4/]

Finally, the SEP review noted several issues where a change in licensing emphasis has occurred since the licensing of Palisades and, therefore, a different or more stringent standard applies today. Design standards and loading combinations are in some instances more stringent now than at the time of the Palisades licensing, due to increased understanding of accident sequences.[5/] The NRC is requiring further study of the Palisades design to address the effect of changes on the adequacy and margins of safety of the structures. Another instance where current criteria are more stringent is the design and monitoring systems for containment isolation. The NRC is requiring the addition of power operated valves, the upgrading of electrical penetration back-up circuit breaker operation, and increased inspection of containment integrity, in order to bring Palisades in line with current criteria.[6/]

3.5.2: SUMMARY OF RESULTS - R. E. GINNA

The main area of difference between current criteria and plant design and construction found at the R. E. Ginna facility was in design topics. Structures at

Ginna may not be adequately designed for the effects of external phenomena such as earthquakes, flooding, or tornadoes, or for loads resulting from accident sequences such as pipe breaks. The standards and models currently applied to determining the adequacy of structures are more stringent than those used in the original review of the R. E. Ginna Plant.

The licensee is performing a major re-evaluation of structures at Ginna, which will encompass the questions raised above.[7/] The NRC is already requiring some flood protection and seismic upgrading at Ginna.[8/]

Accident sequences may disable vital safety equipment at the Ginna plant. To define the potential problems further studies are being required for the following topics:

a) Switchover of Engineered Safety Features from injection to recirculation mode.

b) Auxiliary building tank failures causing flooding of safety equipment.

c) Shutdown cooling in the event of loss of Reactor Heat Removal system due to fire.[9/]

Containment isolation issues arose at Ginna, as well as at Palisades. The NRC will be requiring additional remotely-operated valves and inspection of other valves after pipe break inside containment events to assure containment isolation.[10/]

Finally, two instrumentation changes are being required

by the NRC in order to increase the effectiveness of operators. Monitors and alarms for the DC Power System Bus and for the Component Cooling Water surge tank level are being added to the control room. 11/

3.5.3: UNRESOLVED ISSUES

The SEP objectives identified in Section 1.1 show that the NRC intends to provide a safety review of the design of nuclear power plants which includes all significant safety topics. However, the topics relating to Three Mile Island (TMI) Action Plan items or Unresolved Safety Issues (USI), have been separated from the topic review for the present time. Twenty-four TMI and USI topics have been excluded from SEP review. 12/

The Palisades and R. E. Ginna SEP reports contain further topics which are not resolved because they are related to TMI and USI topics. The six such topics from the Palisades SEP report integrated assessment are:

a) Wind and Tornado Loadings on Safety Injection and Refueling Water, and Condensate Storage Tanks (Topic III-2(1)).

b) Loose Parts Monitoring and Core Barrel Vibration Monitoring (Topic III-8.A).

c) Monitoring of Reactor Coolant Intersystem Leakage (Topic V-5(3)).

d) Removal of Non-essential Loads as an Alternative to GDC 17 (Topic VII-3(2)).

e) Fire Protection of Associated Circuits (Topic IX-6(2)).

f) Failure of Main Feedwater Isolation, Steamline Break (Topic XV-2(2)).

In the R. E. Ginna SEP report, five topics were unresolved because they were related to TMI and USI items. The topics were:

a) Seismic Qualification of Cable Trays (Topic III-6(7)).

b) Loose-Parts Monitoring and Core Barrel Vibration Monitoring (Topic III-8.A).

c) Use of Safety-Grade Systems for Safe Shutdown (Topic V-10.B(2)).

d) Engineering Safety Feature Switchover from Injection to Recirculation Mode (Topic VI-7.B).

e) Fire Protection (Topic IX-6). 14/

TMI and USI related topics were excluded from the SEP review to avoid duplication of effort within the NRC. However, neither the Palisades nor the Ginna SEP report discusses the potential significance of the unresolved issues. Therefore, one cannot judge the relative importance of resolved issues and unresolved issues in the plant-specific SEP reviews.

3.5.4: SUMMARY OF SEP REVIEW RESULTS

Although the SEP integrated assessment reports for Palisades and R. E. Ginna (NUREG-0820 and NUREG-0821, respectively) have been published in draft form, many topics remain

unresolved in both SEP reviews. The topics requiring further study relate mostly to structural adequacy and protection of safety equipment during various accident sequences. Other topics are unresolved because they are related to TMI Action Plan items or USI's.

The SEP reviews have shown that a majority of topics are acceptable by current standards - 57 out of 90 at Palisades and 58 out of 92 at Ginna. The remaining topics were evaluated for safety significance or, in a few cases, corrected on the spot through minor modifications.

Some of the topics evaluated in the integrated assessment phase of the SEP review were accepted on the basis of low safety significance (of the 31 topics in the Palisades integrated assessment, eight were considered acceptable on that basis.) 12/

In summary, the SEP reviews of Palisades and R. E. Ginna have identified the topics of greatest safety significance, although several of those topics are presently unresolved.

SECTION 3

LIST OF REFERENCES

1/ NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant (Draft), Nuclear Regulatory Commission, Washington, D.C., April, 1982, pps. 4-17 to 4-21, 4-41, 4-42

2/ Ibid 1, pps. 4-41 to 4-43.

3/ Ibid 1, pps. 4-43, 4-44.

4/ Ibid 1, pps. 4-21, 4-41 to 4-43.

5/ Ibid 1, pps. 4-23, 4-24.

6/ Ibid 1, pps. 4-33, 4-40, 4-41.

7/ NUREG-0821, Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant (Draft), Nuclear Regulatory Commission, Washington, D.C., May, 1982, pps. 4-1, 4-2, 4-5, 4-6.

8/ Ibid 7, pps. 4-1, 4-2, 4-11, 4-12.

9/ Ibid 7, pps. 4-23 to 4-27.

10/ Ibid 7, pps. 4-10, 4-21 to 4-23.

11/ Ibid 7, pps. 4-24, 4-25.

12/ Ibid 1, Appendix B.

13/ Ibid 1, pp. 4-15, 4-24 to 4-27, 4-36, 4-37, 4-44, 4-47.

14/ Ibid 7, pp. 4-13, 4-15, 4-18, 4-19, 4-23, 4-24, 4-27.

15/ Ibid 1, pp. 4-2 to 4-11.

SECTION 4

COMMENTS ON THE SYSTEMATIC EVALUATION PROGRAM

4.1: INTRODUCTION

This Section summarizes the opinions of several reviewers as to the adequacy of SEP in fulfilling its stated objectives. Section 4.2 sets forth MHB's evaluation of SEP at this time. Comments from the Advisory Committee on Reactor Safeguards (ACRS), the NRC Commissioners, and the NRC Staff's consultants are summarized in Sections 4.3, 4.4, and 4.5, respectively. Section 4.6 contains the conclusions of this MHB report.

4.2: MHB STATUS EVALUATION OF THE SYSTEMATIC EVALUATION PROGRAM

After five years, the SEP has published its first two plant review reports. Eight more reports are expected to follow within the next year. The NRC Staff, Commissioners, and the ACRS have expressed overall satisfaction with the SEP efforts to date. The NRC Staff's consultants gave the SEP program generally favorable evaluations. In general, MHB finds that SEP has made a substantial contribution to evaluating the safety of older plants. Both the methodology and the plant-specific application have been shown to be viable techniques for bridging the gap between early and modern safety evaluation

criteria. However we conclude, as did the ACRS and the NRC Staff's consultants, that the SEP reviews must be considered incomplete, until the Three Mile Island (TMI) Action Plan items and Unresolved Safety Issues (USI) are resolved by the NRC Staff. Twenty-four TMI and USI issues are absent from the SEP review reports. Additionally, several topics are unresolved in the SEP reports because they are related to TMI or USI topics, and several more are unresolved until further studies are performed. Although the NRC's decision to separate the issues makes sense in terms of efficiency, it causes the SEP reports to be incomplete evaluations of overall safety and compliance with the most recent safety standards and criteria.

The remainder of the SEP approach and methodology have been very logical and seem well-suited to the SEP objectives. For example, the listing of all possible safety issues by the Staff, then refining the list to a workable number of topics provides a most straight-forward approach to the creation of a Topic List with reasonable assurance of including dominant contributors.

Backfitting decisions for those topics which were resolved appear to have been based on a combination of engineering judgement, a limited use of Probabilistic Risk Assessment, and a qualitative assessment of the significance of the issue. In general, the discussions accompanying resolution of topics

seem understandable, and appear to give adequate justifications for the Staff decisions. However, at this time, an evaluation of the effectiveness and impact of the SEP decision-making processes and backfit decisions may be premature for two reasons: first, decision must be made on a number of currently unresolved topics, as mentioend above. The decisions on the presently unresolved topics may turn out to be considerably more significant than those decisions which have been made at present. Second, with only two SEP review reports published out of an expected sixty to seventy within the next several years, the backfit decisions made to date may prove to be unrepresentative.

4.3: ACRS COMMENTS ON THE SYSTEMATIC EVALUATION PROGRAM

The ACRS has met twice in general session to discuss the first two SEP reviews: the Palisades and R. E. Ginna nuclear power plant. Letters were sent from the ACRS to the NRC after both meetings summarizing the ACRS position with regard to the plant reviews. The two letters are included herein as Appendix D. The ACRS comments on each report are summarized below.

Palisades Report

The ACRS found the methodology and decision-making approach of the Palisades review to be inadequate. They approved

of the use of a limited PRA in the evaluation of individual topics' safety importance, noting that plant-specific PRA's, when available, may be very helpful input to Integrated Plant Safety Assessments. 1/

The ACRS pointed out that the separation of Three Mile Island (TMI) Action Plan items and Unresolved Safety Issues (USI) makes the final summary in the Palisades report incomplete. 2/ The NRC Staff separated these items in order to avoid duplication of effort. However, as mentioned in Section 1.1, objective #4 of the SEP program is to "assess the safety adequacy of the design and operation of currently licensed nuclear plants". The ACRS considers the SEP reviews to fulfill objective #4 in part, rather than completely.

Finally, the ACRS noted that further information such as calculations, studies, and evaluations remain to be submitted by the licensee for nine topics, and therefore the ACRS endorsement of the Palisades report was based on a review of only those topics which were completed. 3/

R. E. Ginna Report

The ACRS letter on the Ginna SEP Review restated the comments first voiced in the Palisades review, on use of PRA data, separation of TMI and USI topics, and the general adequacy of the SEP review methodology. For Ginna, the ACRS noted

that seven topics were incomplete and requiring further study by the licensee. 4/

The ACRS also commented on three issues for which the licensee disagreed with the NRC Staff resolution. The ACRS supported the Staff's position in each case. 5/ These issues were: setting a conservative groundwater level, whether to study the effect of flooding from a nearby creek, and whether to upgrade isolation valves.

4.4: NRC COMMISSIONERS' COMMENTS ON THE SYSTEMATIC EVALUATION PROGRAM

The NRC Commissioners were briefed by members of the NRC Staff as to the status of SEP in October 1981: The Commissioners were unable to evaluate the reports on the Palisades and R. E. Ginna plants, because the reports had not been completed at the time. However, the Systematic Evaluation Program in general was discussed, and the following opinions could be identified from the discussion:

1. The Program has been a worthwhile and important effort. This opinion was voiced by virtually all of the Commissioners. Chairman Palladino added that although it has taken many years to produce tangible results, SEP will be an important effort both for the first group of eleven plants, and for the plants which will be reviewed in the future. 6/

2. The NRC Staff should review the experience of the Federal Aviation Agency and the Environmental Protection Agency, both of

which had to consider the costs and benefits of requiring major backfitting by regulated industries. The suggestior was made by Chairman Palladino, and Commissioners Gilinski and Bradford agreed. 7/

3. The NRC Staff should identify the plants for which there is most concern, and review those as soon as possible. The suggestion was made by Commissioner Gilinski and Chairman Palladino agreed. However, there was little consensus as to which plants should be identified in the "most concern" category. Commissioner Gilinsky suggested that the first generation of large plants, especially those sited near population centers, should be reviewed next by SEP. 8/ Later, he suggested that some plants, licensed in 1973 and 1974, were reviewed very quickly and perhaps not thoroughly. 9/ A representative of the NRC Staff said that the main area of concern was not necessarily a group of plants, but how well NRC reviewers understood the requirements and their implications at a given time. 10/

4. The Commissioners seemed to approve of the NRC Staff's plan to consult with former NRC Staff and Commissioners to get objective critiques of the SEP results. 11/ The consultants' evaluations of SEP results are discussed in Section 4.5.

4.5: NRC STAFF CONSULTANTS COMMENTS ON THE SYSTEMATIC EVALUATION PROGRAM

In April 1982, five NRC Staff consultants reviewed the SEP program in general and the SEP Palisades report in particular.* The consultants were Drs. Robert Budnitz, Stephen Bush, Joseph Hendrie, Herbert Isbin, and Zenon Zudans.

All of the consultants considered the concept, methodology, and decision-making approach of SEP to be excellent and in keeping with the program objectives. Each qualified his endorsement by noting that the TMI Action Plan Items, and

* NRC Staff Consultants Reviews are included herein as Appendix E.

Unresolved Safety Issues, had not been addressed in the SEP report. The consultants' conclusions as to the general value of the program were not unanimous. Two consultants, Drs. Hendrie and Zudan, apparently feel that the program to date has succeeded in its goals. Drs. Bush and Isbin gave mixed reviews on the question of overall value, and Dr. Budnitz apparently felt that the SEP review did not focus on "the real issues of safe operation of Palisades". [12/] Three consultants commented upon the review of operating experience in the SEP report on Palisades. Dr. Isbin suggested that it be updated and expanded to include more indicators of licensee performance. [13/] Drs. Budnitz and Bush pointed out that no evaluation of management and in-house engineering competence and responsiveness were included in the SEP review, although these items have significant safety impact. [14/][15/] The following excerpts from the consultants' reports give an indication of how the individual reviewers felt about SEP:

Dr. Hendrie -

> "It is going on 5 years since the SEP Phase II came before the Commission for approval. I voted for it with a certain amount of trepidation. I had some concern then over the staff's ability to do a balanced assessment on an older operating plant and to come up with results that were meaningful from a safety standpoint and did not simply end up requiring total conformance with current criteria regardless of the safety benefits. That concern did not abate much in the years following and I used to confront the bright-

eyed proposers of an SEP Phase III with the direction to go back and produce something from Phase II and then we would see. Now we have the first product from Phase II. I think it is a good job. My compliments to the staff." 16/

Dr. Bush -

"With regard to equipment and design items, the authors addressed to a major degree the SEP task force objectives as well as applying the tiered criteria to resolve deviations. Generally, the approach is even-handled, not requiring backfit arbitrarily. I am less satisfied with the handling of operating history.

Appendix F points out the high incidence of loss of power. This combined with some of the operator errors listed could yield a definite degradation in safety margins." 17/

Dr. Budnitz -

"I will summarize by stating that I believe this first SEP report has been quite successful: the metaphor of the laundry list that has been cleaned up is appropriate. Maybe Palisades can get a regular operating license now for one thing; and maybe the utility staff and the regulatory staff can go on from this mop-up activity to think hard about the real issues of safe operation of Palisades, issues hardly dealt with in the analyses within this draft report.

Finally, I do think it is important to state my view that it is only in retrospect, after the analysis, that one is at liberty to characterize the SEP list for Palisades as a 'laundry list': beforehand, we didn't really know what would crop up. So in that regard the activity has been successful indeed." 18/

Dr. Isbin -

"The highlights of my review are as follows:

- The planning used for SEP is outstanding from the point of view of identifying safety items.

- The objectives have been well conceived; however one major objective may have been inadvertently omitted in the NUREG report.

- The review of operating experiences needs to be updated and augmented.

- Limited assistance was provided by the probabilistic risk assessment for this plant.

- The reporting of the Topics and the ensuing approach to the decision making, in general, are well done.

- Too many events and changes have occurred in the past three years to be able to evaluate whether the SEP program is efficiently and economically using NRC and Industry resources.

- An important finding is that no SEP Topics was considered to be of sufficient importance to require a prompt resolution.

- Attention has been focussed on achieving an '...integrated and balanced...' decision, considering that the SEP program is being carried out in conjunction with major NRC and Industry efforts for implementing TMI Action Plan Items, and responding to IE Bulletins and Generic Letters. Resolution of Unresolved Safety Issues remains a continuing activity along with mandated annual reports to the Congress concerning identification of any new issues. Overall assessment of safety of the plant must utilize all these inputs." 19/

Dr. Zudans -

"My overall impression of Palisades SEP Review is that considerably more sound engineering effort has been put in Palisades SEP review, in particular in terms of proper understanding of design, processes and consequences involved, than maybe normally done

> during regular licensing review process, (SEP Topic list covers essentially all safety related design aspect of a nuclear power plant). The process however, is not complete until all open items are resolved in an integrated manner." 20/

4.6: STUDY CONCLUSIONS

The SEP reviews have resolved, for the Palisades and R. E. Ginna nuclear plants, a large number of topics which the reviews have shown to have little or no safety significance. The reviews have also resolved several other topics through requiring modifications or changes to Technical Specifications at the plants.

The remaining topics appear to be the most significant. They are Three Mile Island (TMI) Action Plan items, Unresolved Safety Issues (USI), or SEP topics requiring further studies. All of the reviewers of SEP have commented on the absence of TMI and USI topics from the SEP integrated assessments. Although opinions varied as to the significance of the unresolved issues, it is clear that the SEP reviews of nuclear plants will not be complete safety reviews until the TMI and USI topics are integrated into the safety assessments.

Two further aspects of the Systematic Evaluation Program are expected to develop within the next year. First, during the remainder of the SEP Phase II reviews, the NRC Staff intends to identify those topics which are consistently

of low safety significance, and can be dropped from the SEP reviews. Second, after Phase II reviews are complete, the NRC Staff expects to begin Phase III SEP reviews. The Phase III reviews are expected to be more efficient, using the lessons learned from the Phase II SEP reviews. Whether the current problems with the SEP reviews (e.g. separation of USI and TMI issues) can be resolved within the next year is unknown. If not, the same problems will exist with Phase III SEP reviews as exist now.

SECTION 4

LIST OF REFERENCES

1/ ACRS letter to NRC, May 11, 1982, p. 3.

2/ Ibid 1, p. 2.

3/ Ibid 1, p. 3.

4/ ACRS letter to NRC, August 18, 1982, p. 2.

5/ Ibid 4, pp. 2-3.

6/ Transcript of October 22, 1981 Commission Meeting, U. S. Nuclear Regulatory Commission, Washington, D.C., p. 86.

7/ Ibid 6, pp. 67-69.

8/ Ibid 6, pp. 70-71.

9/ Ibid 6, p. 75.

10/ Ibid 6, p. 75.

11/ Ibid 6, pp. 68-69.

12/ Dr. Robert Budnitz letter to William Russell, Office of Nuclear Reactor Regulation, April 15, 1982, (See Appendix E), p. 6.

13/ Dr. H. S. Isbin letter to Project Officer W. Russell, April 23, 1982 (see Appendix E), pp. 1-2.

14/ Ibid 12, p. 5.

15/ "Palisades Plant - A Critique of the Integrated Plant Safety Assessment, Systematic Evaluation Program", S. H. Bush, (see Appendix E), p. 3.

16/ Dr. Joseph Hendrie letter to William Russell, Systematic Evaluation Program Branch, April 27, 1982 (see Appendix E), p. 8.

17/ Ibid 15, pp. 5-6.

18/ Ibid 12, p. 6.

19/ Ibid 13, pp. 1-2, summary.

20/ Dr. Zenon Zudans letter to William Russell, SEP Project Manager/Technical Coordinator, April 28, 1982 (see Appendix E), p. 2.

SECTION 5

BIBLIOGRAPHY

5.1: BIBLIOGRAPHY

The MHB SEP review has included a large volume of reports, memoranda, and other documents. The most relevant and informative documents are listed below for the benefit of the reader:

1. NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant, U.S. Nuclear Regulatory Commission, Washington, D.C., April, 1982.

2. NUREG-0821, Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant, U.S. Nuclear Regulatory Commission, Washington, D.C., May, 1982.

3. Transcript of April 15, 1982 Advisory Committee on Reactor Safeguards Subcommittee on the Systematic Evaluation Program meeting, U.S. Nuclear Regulatory Commission, ACRS, Washington, D.C.

4. Transcript of June 30, 1982 Advisory Committee on Reactor Safeguards Subcommittee on the Systematic Evaluation Program meeting, U.S. Nuclear Regulatory Commission, ACRS, Washington, D.C.

5. Transcript of May 7, 1982 Advisory Committee on Reactor Safeguards General Meeting, U.S. Nuclear Regulatory Commission, ACRS, Washington, D.C.

6. Transcript of July 9, 1982 Advisory Committee on Reactor Safeguards General Meeting, U.S. Nuclear Regulatory Commission, ACRS, Washington, D.C.

7. ACRS letter to NRC, May 11, 1982.

8. ACRS letter to NRC, August 18, 1982.

9. Transcript of January 14, 1981 Commission Meeting, U.S. Nuclear Regulatory Commission, Washington, D.C.

10. Transcript of October 22, 1981 Commission Meeting, U.S. Nuclear Regulatory Commission, Washington, D.C.

11. Dr. Robert Budnitz letter to William Russell, Office of Nuclear Reactor Regulation, April 15, 1982.

12. Dr. H. S. Isbin letter to Project Officer W. Russell, April 23, 1982.

13. "Palisades Plant - A Critique of the Integrated Plant Safety Assessment, Systematic Evaluation Program," S. H. Bush.

14. Dr. Joseph Hendrie letter to William Russell, Systematic Evaluation Program Branch, April 27, 1982.

15. Dr. Zenon Zudans letter to William Russell, SEP Project Manager/Technical Coordinator, April 28, 1982.

16. NUREG-0485, Systematic Evaluation Program, Status Summary Report, U.S. Nuclear Regulatory Commission, Washington, D.C. (published monthly)

17. NRC letter to SEP licensees, January 14, 1981, U.S. Nuclear Regulatory Commission, Washington, D.C.

18. SECY-77-561, Systematic Evaluation of Operating Reactors, Phases I and II, memorandum for the Commissioners, U.S. Nuclear Regulatory Commission, Washington, D.C., October 26, 1977.

19. SECY-80-414, Systematic Safety Evaluation of All Currently Operating Nuclear Power Reactors, memorandum for the Commissioners, U.S. Nuclear Regulatory Commission, Washington, D.C., September 9, 1980.

20. SECY-81-13, Systematic Safety Evaluation of All Currently Operating Nuclear Power Reactors, memorandum for the Commissioners, U.S. Nuclear Regulatory Commission, Washington, D.C., January 8, 1981.

21. Office of Nuclear Reactor Regulation Report on the Systematic Evaluation of Operating Facilities, U.S. Nuclear Regulatory Commission, Washington, D.C., November 25, 1977.

LIST OF APPENDICES

APPENDIX	DESCRIPTION
A	SYSTEMATIC EVALUATION PROGRAM LIST OF 137 TOPICS
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C	MODIFICATIONS REQUIRED BY THE NRC IN THE SEP REVIEWS OF PALISADES AND R. E. GINNA PLANTS
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E	NRC STAFF CONSULTANTS' REVIEWS OF PALISADES PLANT SEP REPORT (NUREG-0820)
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APPENDIX A

SYSTEMATIC EVALUATION PROGRAM

LIST OF 137 TOPICS

(Source: NUREG-0485, SEP Status
Summary Report, August 31, 1982)

SYSTEMATIC EVALUATION PROGRAM TOPIC LIST

TOPIC #	TITLE
II-1	A. Exclusion Area Authority and Control
	B. Population Distribution
	. Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional Industrial, and Military Facilities
II-2	A. Severe Weather Phenomena
	*B. Onsite Meteorological Measurements Program
	C. Atmospheric Transport and Diffusion Characteristics for Accident Analysis
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II-3	. Hydrologic Description
	B. Flooding Potential and Protection Requirements
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	C. Safety-related Water Supply (Ultimate Heat Sink (UHS))
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III-2	Wind and Tornado Loadings
III-3	A. Effects of High Water Level on Structures
	B. Structural and Other Consequences (e.g. Flooding of Safety-Related Equipment in Basements) of Failure of Underdrain Systems
	C. Inservice Inspection of Water Control Structures
III-4	A. Tornado Missiles
	B. Turbine Missiles
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III-5	A. Effects of Pipe Break on Structures, Systems and Components Inside Containment
	B. Pipe Break Outside Containment
III-6	Seismic Design Considerations
III-7	A. Inservice Inspection, Including Prestressed Concrete Containments with Either Grouted or Ungrouted Tendons
	B. Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria
	C. Delamination of Prestressed Concrete Containment Structures
	D. Containment Structural Integrity Tests
III-8	A. Loose Parts Monitoring and Core Barrel Vibration Monitoring
	B. Control Rod Drive Mechanism Integrity
	C. Irradiation Damage, Use of Sensitized* Stainless Steel and Fatigue Resistance
	*D. Core Supports and Fuel Integrity
III-9*	Support Integrity
III-10	A. Thermal-Overload Protection for Motors of Motor-operated Valves
	B. Pump Flywheel Integrity
	C. Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves
III-11*	Component Integrity
III-12*	Environmental Qualification of Safety Related Equipment
IV-1	A. Operation with Less Than All Loops In-Service
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures
*IV-3	BWR Jet Pumps Operating Indications
V-1	Compliance with Codes and Standards (10 CFR 50.55a)
V-2	Applicable Code Cases
V-3*	Overpressurization Protection
V-4*	Piping and Safe End Integrity
V-5	Reactor Coolant Pressure Boundary (RCPB) Leakage, Detection
V-6	Reactor Vessel Integrity
V-7	Reactor Coolant Pump Overspeed
V-8*	Steam Generator (SG) Integrity
V-9	Reactor Core Isolation Cooling System (BWR)
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	B. RHR Reliability
V-11	A. Requirements for Isolation of High and Low Pressure Systems
	B. RHR Interlock Requirements
V-12	A. Water Purity of Boiling Water Reactor Primary Coolant
V-13*	Water Hammer
VI-1	Organic Materials and Post Accident Chemistry
VI-2	A. Pressure-Suppression Type BWR Containments
	*B. Subcompartment Analysis
	C. Ice Condenser Containment
	D. Mass and Energy Release for Possible Pipe Break Inside Containment
VI-3	Containment Pressure and Heat Removal Capability
VI-4	Containment Isolation System
VI-5*	Combustible Gas Control
VI-6	Containment Leak Testing

SYSTEMATIC EVALUATION PROGRAM TOPIC LIST

TOPIC #	TITLE
VI-7.A	1. ECCS Re-evaluation to Account for Increased Vessel Head Temperature
	2. Upper Plenum Injection
	3. ECCS Actuation System
	4. Core Spray Nozzle Effectiveness
VI-7.B	ESF Switchover from Injection to Recirculation Mode (Automatic ECCS Realignment)
VI-7.C	ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves
	1. Appendix K - Electrical Instrumentation and Control (EIC) Re-reviews
	2. Failure Mode Analysis - ECCS
	3. The Effect of PWR Loop Isolation Valve Closure During a LOCA on ECCS Performance
VI-7.D	Long Term Cooling - Passive Failures (e.g., Flooding of Redundant Components)
VI-7.E*	ECCS Sump Design and Test for Recirculation Mode Effectiveness
VI-7.F	Accumulator Isolation Valves Power and Control System Design
VI-8*	Control Room Habitability
VI-9	Main Steam Line Isolation Seal System - BWR
VI-10	A. Testing of Reactor Trip System and Engineered Safety Features Including Response Time Testing
	B. Shared Engineered Safety Features, On-site Emergency Power, and Service Systems for Multiple Unit Facilities
VII-1	A. Isolation of Reactor Protection System from Non-Safety Systems, Including Qualifications of Isolation Devices
	B. Trip Uncertainty and Setpoint Analysis Review of Operating Data Base
VII-2	Engineered Safety Features (ESF) System Control Logic and Design
VII-3	Systems Required for Safe Shutdown
VII-4*	Effects of Failure in Non-Safety Related Systems on Selected Engineered Safety Features
VII-5*	Instruments for Monitoring Radiation and Process Variables During Accidents
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VIII-1	A. Potential Equipment Failures Associated with a Degraded Grid Voltage
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IX-2*	Overhead Handling Systems - Cranes
IX-3	Station Service and Cooling Water Systems
IX-4	Boron Addition System
IX-5	Ventilation Systems
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XV-2	Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)
XV-3	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
XV-4	Loss of Non-Emergency A-C Power to the Station Auxiliaries
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XV-7	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
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XV-9	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
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XV-12	Spectrum of Rod Ejection Accidents (PWR)
XV-13	Spectrum of Rod Drop Accidents (BWR)*
XV-14	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory
XV-15	Inadvertent Opening of a PWR Pressurizer Safety-Relief Valve or a BWR Safety/Relief Valve
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-17	Steam Generator Tube Failure (PWR)
XV-18	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)
XV-19	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
XV-20	Radiological Consequences of Fuel Damaging Accidents (Inside and Outside Containment)
XV-21*	Spent Fuel Cask Drop Accidents
XV-22*	Anticipate Transients Without Scram
XV-23*	Multiple Tube Failures in Steam Generators
XV-24*	Loss of All A-C Power
XVI	Technical Specifications
XVII	Operational QA Program

*Topic deleted from SEP program due to duplication by Three Mile Island (TMI) NRC Action Plan, Unresolved Safety Issue or other SEP Topic per May 7, 1981 letter from G. Lainas to all SEP Licensees.

APPENDIX B

TOPICS EVALUATED AS ACCEPTABLE
IN SEP REVIEWS
OF PALISADES AND R. E. GINNA PLANTS

PALISADES PLANT

(Source: NUREG-0820, Integrated Plant
Safety Assessment, SEP,
Palisades Plant)

3.2 Topics for Which Plant Design Meets Current Criteria or Was Acceptable on Another Defined Basis

TOPIC	TITLE
II-1.B	Population Distribution
II.1-C	Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities
II-2.A	Severe Weather Phenomena
II.2-C	Atmospheric Transport and Diffusion Characteristics for Accident Analysis
II.3-A	Hydrologic Description
II-4	Geology and Seismology
II-4.A	Tectonic Province
II-4.B	Proximity of Capable Tectonic Structures in Plant Vicinity
II-4.C	Historical Seismicity Within 200 Miles of Plant
II-4.D	Stability of Slopes
II-4.F	Settlement of Foundations and Buried Equipment
III-3.A	Effects of High Water Level on Structures
III-4.B	Turbine Missiles
III-4.C	Internally Generated Missiles
III-4.D	Site-Proximity Missiles (Including Aircraft)
III-5.B	Pipe Break Outside Containment
III-7.D	Containment Structural Integrity Tests
III-8.C	Irradiation Damage, Use of Sensitized Stainless Steel, and Fatigue Resistance
III-10.A	Thermal-Overload Protection for Motors of Motor-Operated Valves
III-10.B	Pump Flywheel Integrity
IV-1.A	Operation With Less Than All Loops In Service

TOPIC	TITLE
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures
V-6	Reactor Vessel Integrity
V-7	Reactor Coolant Pump Overspeed
V-10.A	Residual Heat Removal System Heat Exchanger Tube Failures
V-10.B	Residual Heat Removal System Reliability
VI-1	Organic Materials and Postaccident Chemistry
VI-7.A.3	Emergency Core Cooling System Actuation System
VI-7.B	Engineered Safety Feature Switchover From Injection to Recirculation Mode (Automatic Emergency Core Cooling System Realignment)
VI-7.C	Emergency Core Cooling System (ECCS) Single-Failure Criterion and Requirements for Locking Out Power to Valves, Including Independence of Interlocks on ECCS Valves.
VI.7.C.1	Appendix K - Electrical Instrumentation and Control Re-reviews
VI-7.C.2	Failure Mode Analysis (Emergency Core Cooling System)
VI-7.D	Long-Term Cooling Passive Failures (e.g., Flooding of Redundant Components)
VI-7.F	Accumulator Isolation Valves Power and Control System Design
VII-1.B	Trip Uncertainty and Setpoint Analysis Review of Operating Data Base
VII-2	Engineered Safety Feature System Control Logic and Design
VII-6	Frequency Decay
VIII-1.A	Potential Equipment Failures Associated With Degraded Grid Voltage
VIII-2	Onsite Emergency Power Systems (Diesel Generator)
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation
IX-1	Fuel Storage
IX-4	Boron Addition System (PWR)
XIII-2	Safeguards/Industrial Security

TOPIC	TITLE
XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve
XV-3	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
XV-4	Loss of Nonemergency AC Power to the Station Auxiliaries
XV-5	Loss of Normal Feedwater Flow
XV-6	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
XV-7	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
XV-8	Control Rod Misoperation (System Malfunction or Operator Error)
XV-9	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
XV-10	Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)
XV-14	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory
XV-15	Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-17	Radiological Consequences of Steam Generator Tube Failure (PWR)
XV-19	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
XV-20	Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment)
XVII	Operational Quality Assurance Program

R. G. GINNA PLANT

(Source: NUREG-0821, Integrated Plant
Safety Assessment,
SEP, R. E. Ginna Plant)

TOPICS FOR WHICH PLANT DESIGN MEETS CURRENT CRITERIA OR WAS ACCEPTABLE ON ANOTHER DEFINED BASIS

TOPIC	TITLE
II-1.B	Population Distribution
II-1.C	Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities
II-2.C	Atmospheric Transport and Diffusion Characteristics for Accident Analysis
II-3.A	Hydrologic Description
II-4	Geology and Seismology
II-4.A	Tectonic Province
II-4.B	Proximity of Capable Tectonic Structures in Plant Vicinity
II-4.C	Historical Seismicity Within 200 Miles of Plant
II-4.F	Settlement of Foundations and Buried Equipment
III-4.B	Turbine Missiles
III-4.D	Site-Proximity Missiles (Including Aircraft)
III-7.C	Delamination of Prestressed Concrete Containment Structures
III-7.D	Containment Structural Integrity Tests
III-8.C	Irradiation Damage, Use of Sensitized Stainless Steel, and Fatigue Resistance
III-10.A	Thermal-Overload Protection for Motors of Motor-Operated Valves

TOPIC	TITLE
III.10.B	Pump Flywheel Integrity
IV-1.A	Operation with Less Than All Loops in Service
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures
V-6	Reactor Vessel Integrity
V-7	Reactor Coolant Pump Overspeed
V-11.A	Requirements for Isolation of High- and Low-Pressure Systems
V-11.B	Residual Heat Removal System Interlock Requirements
VI-1	Organic Materials and Postaccident Chemistry
VI-2.D	Mass and Energy Release for Postulated Pipe Break Inside Containment
VI-3	Containment Pressure and Heat Removal Capability
VI-6	Containment Leak Testing
VI-7.A.1	Emergency Core Cooling System Reevaluation To Account for Increased Vessel Upper-Head Temperature
VI-7.A.2	Upper Plenum Injection
VI-7.A.3	Emergency Core Cooling System Actuation System
VI-7.C	Emergency Core Cooling System (ECCS) Single-Failure Criterion and Requirements for Locking Out Power to Valves, Including Independence of Interlocks on ECCS Valves
VI-7.C.1	Appendix K--Electrical Instrumentation and Control Re-reviews

TOPIC	TITLE
VI-7.C.2	Failure Mode Analysis (Emergency Core Cooling System)
VI-7.D	Long-Term Cooling Passive Failures (e.g., Flooding of Redundant Components)
VI-7.F	Accumulator Isolation Valves Power and Control System Design
VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing
VII-1.A	Isolation of Reactor Protection System from Nonsafety Systems, Including Qualification of Isolation Devices
VII-1.B	Trip Uncertainty and Setpoint Analysis Review of Operating Data Base
VII-2	Engineered Safety Features System Control Logic and Design
VII-3	Systems Required for Safe Shutdown
VII-6	Frequency Decay
VIII-1.A	Potential Equipment Failures Associated With Degraded Grid Voltage
VIII-2	Onsite Emergency Power System (Diesel Generator)
VIII-3.A	Station Battery Capacity Test Requirements
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation
VIII-4	Electrical Penetrations of Reactor Containment
IX-1	Fuel Storage

TOPIC	TITLE
IX-4	Boron Addition System (PWR)
XIII-2	Safeguards/Industrial Security
XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve
XV-2	Spectrum of Steam System Piping Failures Inside and Outside Containment (PWR)
XV-3	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
XV-4	Loss of Nonemergency AC Power to the Station Auxiliaries
XV-5	Loss of Normal Feedwater Flow
XV-6	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)
XV-7	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
XV-8	Control Rod Misoperation (System Malfunction or Operator Error)
XV-9	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
XV-10	Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)

TOPIC	TITLE
XV-12	Spectrum of Rod Ejection Accidents (PWR)
XV-14	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory.
XV-15	Inadvertent Opening of a PWR Pressurizer Safety/ Relief Valve or a BWR Safety/Relief Valve
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-17	Radiological Consequence of Steam Generator Tube Failure (PWR)
XV-19	Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
XV-20	Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment)
XVII	Operational Quality Assurance Program

APPENDIX C

MODIFICATIONS REQUIRED BY NRC IN THE SEP REVIEWS OF PALISADES AND R. E. GINNA PLANTS

PALISADES PLANT

(Source: NUREG-0820, Integrated Plant
Safety Assessment, SEP,
Palisades Plant)

SAFETY IMPROVEMENTS IMPLEMENTED DURING TOPIC REVIEW

(1) The anchorage and support systems of all safety-related electrical equipment were evaluated and were upgraded as needed (3.3.1).

(2) Bracing was provided to prevent lateral movement of the diesel generator oil tank (3.3.1).

(3) A seismic reanalysis of all safety-related piping systems more than 2-1/2 in. in diameter was conducted. The required modifications of piping systems and supports were completed (3.3.1).

(4) New batteries, which have a nominal capacity of 2 hours, were installed to replace 30-minute batteries (3.3.2).

(5) Separate annunciators were installed for the following diesel generator functions: (a) control switch not in automatic, (b) overspeed, (c) overcrank, and (d) loss of dc control (3.3.3).

(6) The following new alarms were installed to indicate the dc power system operability status: (a) 125-vdc tie breaker open (both buses) (b) public address system inverter loss of voltage (bus D10 only), and (c) battery undervoltage (both batteries) (3.3.4).

(7) As a result of recent system modifications made to penetrations 17, 17a, 21, 21a, 28, 29, 48, and 73, threaded pipe caps are no longer used as the first boundary to containment isolation (3.3.5).

SAFETY IMPROVEMENTS TO BE IMPLEMENTED BY THE LICENSEE AS A RESULT OF SEP

These improvements fall into two categories. The first category is the group of changes or engineering analyses that the licensee has agreed to make and that will be required by the NRC. The second category is the group of changes that become part of the operating license in the form of Technical Specification changes. Both of these categories are listed below, and the issues are discussed in sections of this report given in parentheses.

Required by NRC

(1) Perform analysis of flooding level caused by seiche (4.2, 4.3, 4.4).

(2) Analyze and upgrade, if necessary, structures, components, and systems (4.5).

(3) Develop procedures to achieve cold shutdown using alternate sources of water and safety-grade equipment if preferred sources of water are not available (4.6.1, 4.8.1, 4.16.2, 4.24.5).

(4) Analyze and provide protection, if necessary, for approximately 200 pipe-break locations inside containment (4.9).

(5) Evaluate safety-related piping less than 2-1/2 in. in diameter under postulated seismic loads (4.10.1).

(6) Evaluate safety-related electrical components under postulated siesmic loads (4.10.2).

(7) Assess structural code changes on safety margins in "as-built" structures (4.12).

(8) Change manual isolation valve on penetration 44 to a power-operated valve (4.20.3).

(9) Weld pipe joints on 3-in. pressurization line of penetration 19 (4.20.4).

(10) Install qualified isolation devices on the steam generator A and B pressure channels and on the reactor coolant flow channel going to the plant computer (4.23.1).

(11) Develop procedure to remove nonessential loads from the battery if the immediate sources of offsite and onsite power are not available. (4.24.1).

(12) Install another level sensor to the component cooling water surge tank and its indicator in the control room (4.24.2).

(13) Complete analysis and modify, if necessary, electrical penetration circuit overload protection (4.26).

(14) Demonstrate adequate service water flow for postulated diesel failures or develop procedures to ensure adequate service water flow distribution (4.27.1).

(15) Provide adequate flood protection for safety systems in the intake structures (4.27.2).

(16) Demonstrate, by test or analysis, the operability of the auxiliary feedwater pumps with a loss of ventilation (4.28.1).

(17) Demonstrate that equipment serviced in the cable-spreading, switchgear, and battery rooms would not be adversely affected by lack of ventilation, or provide proposed system modifications (4.28.2).

Technical Specification Changes

(1) Develop tendon force acceptance criteria for each tendon that vary with time (4.11.1).

(2) Develop Technical Specifications concerning the operability of leak detection systems. This will depend on the outcome of Topic III-5.A (4.15.2).

(3) Place the overpressurization protection system in service before the shutdown cooling system (4.16.1).

(4) Require verification of check-valve closure before criticality after each use of the low-pressure safety injection system for shutdown cooling (4.17).

(5) Verify airlock door seal integrity within 72 hours of each opening or the first of a series of openings, whenever containment integrity is required (4.21).

(6) Conduct battery tests consisting of a battery service test and a battery discharge test to ensure that the battery has sufficient capacity (4.25).

(7) Perform appropriate modifications to make the main steam isolation valve/main steam configuration single-failure-proof with respect to two-steam generator blowdown inside containment (4.30.1).

OTHER SAFETY IMPROVEMENTS REQUIRED BY THE STAFF AS A RESULT OF SEP

The staff has determined that the following improvements are required. The licensee has either not responded to or disagrees with the position taken by the staff. The issues are discussed in the sections of this report given in parentheses. The status of review or the position taken by the licensee is also given.

(1) Perform inservice inspection of water control structures (4.7.1; licensee disagrees with part of the position taken by the staff).

(2) Inspect the concrete surrounding the tendon-end anchorages under integrated leak rate test load if new cracks are noted around the anchorage during the tendon inservice inspection (4.11.3; licensee disagrees with position taken by the staff).

(3) Inspect for dome delamination, using the method described in SER dated May 21, 1981, by July 30, 1982 and every 5 years thereafter (4.13; licensee disagrees with position taken by the staff).

(4) Provide alternative instrumentation for fire protection (4.29.1; licensee has been requested to provide additional information within 60 days of receipt of the draft SER).

R. E. GINNA PLANT

Source: NUREG-0821, Integrated Plant Safety Assessment, SEP, R. E. Ginna Plant)

SAFETY IMPROVEMENTS IMPLEMENTED DURING TOPIC REVIEW

Safety improvements that have been implemented as a result of SEP topic review are listed below and are discussed in sections of this report given in parentheses. Some of these modifications resulted from reviews that were concurrent with the Systematic Evaluation Program.

(1) A check valve leak testing program has been added to the Technical Specifications for check valves in low-pressure systems interfacing with the reactor coolant system (3.3.3.1).

(2) The doorway between the mechanical equipment room and the battery rooms has been replaced with a watertight wall. If a service water line should break, a water relief valve has been provided in the mechanical equipment room that relieves into the turbine building (3.3.1.3).

(3) The anchorage and support system (internal and external) of all safety-related electrical equipment has been upgraded (3.3.2).

(4) The anchorage and support system of the battery racks has been upgraded (3.3.2).

(5) The override capability for the individual containment isolation valves has been modified so that

   (a) manual isolation is never bypassed

   (b) only the high containment radiation or safety injection signal, but not both, is bypassed by a single operator action of a reset button (3.3.4).

(6) A protective tube for the safety injection reset button has been installed (3.3.4).

(7) A containment isolation valve reset panel and relay cabinets have been installed so that containment isolation reset no longer allows valves to return to their preaccident condition (3.3.4).

(8) Setpoints, calibrations, and surveillance requirements associated with the undervoltage protection have been added to the Technical Specifications (3.3.7).

(9) Undervoltage relay racks have been installed in response to a generic letter on grid voltage degradation dated June 3, 1977 (3.3.7).

(10) The delay timer for the sodium hydroxide addition valves has been set to 1 second. Also, operating procedures have been modified so that operators will not override sodium hydroxide addition in the event of a containment spray activation (3.3.11).

(11) Seismic reanalysis of the turbine building bracing has been performed (4.15.1).

SAFETY IMPROVEMENTS TO BE IMPLEMENTED BY THE LICENSEE AS A RESULT OF SEP

These improvements fall into two categories. The first category is the group of changes or engineering analyses to be implemented by the licensee. The second category is the group of changes that will become part of the operating license in the form of Technical Specification changes. Both of these categories are listed below, and the issues are discussed in sections of this report given in parentheses. The licensee has agreed to all of these improvements.

Design or Analysis Requirements

(1) Install a hose connection for the use of the fire water system as an alternate cooling water source for the diesel generators (3.3.1.1).

(2) Provide jet impingement shielding in the intermediate building in the vicinity of the main steamlines and protection for the power-operated relief valves and safety valves from the effects of failure of the turbine building wall (3.3.1.2).

(3) Install a redundant refueling water storage tank level transmitter and indicator (3.3.5).

(4) Develop an instrument string response-time testing program for the auxiliary feedwater and the containment isolation instrumentation (3.3.6).

(5) Provide the necessary circuit protection to ensure that electrical penetrations conform to current licensing criteria (3.3.9).

(6) Perform modifications necessary for implementation of the fire protection dedicated shutdown system (3.3.12).

(7) Analyze and provide structural upgrading as necessary for various loads and load combinations (4.2, 4.8, 4.11, and 4.17).

(8) Implement modifications to the inservice inspection program for water control structures, as required by the staff (4.10.1).

(9) Analyze and provide protection, if necessary, for various safety-related equipment from tornado missiles (4.12).

(10) Provide restraints to the operator of valve CV 5738 to preclude missile generation (4.12.3).

(11) Analyze and provide protection, if necessary, for five pipe-break locations inside containment (4.13).

(12) Provide protection for electrical buses in screenhouse from pipe breaks outside containment (4.14).

(13) Upgrade auxiliary building bracing (4.15.1).

(14) Upgrade structural and functional integrity of essential service water pumps (4.15.3).

(15) Analyze and upgrade, if necessary, three safety-related tanks (4.15.4).

(16) Demonstrate by test or analysis and upgrade, if necessary, the structural integrity of the main control board (4.15.5).

(17) Assess structural code changes on safety margins in "as-built" structures (4.17).

(18) Perform containment liner analysis (4.17.1).

(19) Develop procedures to achieve cold shutdown using safety-grade systems (4.21.2).

(20) Analyze and modify, if necessary, the procedures for the engineered safety feature switchover from injection to recirculation (4.23.1).

(21) Install local indication for battery current, charger output current, and battery high discharge rate with an alarm to be annunciated in the control room on the status of the dc power system (4.24).

(22) Install redundant component cooling water surge tank level indicator and alarms (4.25.2).

(23) Analyze and upgrade, if necessary, tanks in the auxiliary building whose failure would flood out safety-related equipment in the lower levels (4.25.3).

(24) Demonstrate that on a loss of residual heat removal, the solid steam generator system with controlled steam generator blowdown is a viable alternate method for cooldown (4.27).

(25) Analyze and modify the power supply to the fire detection and suppression system (4.27).

Technical Specifications Changes

Some Technical Specifications were implemented during the topic review. However, most of these Technical Specification changes are commitments that have not been submitted. The staff's position is that these proposed Technical Specification changes may be submitted all together following the completion of the integrated assessment. The licensee should submit within 90 days after the issuance of the Final Integrated Plant Safety Assessment Report a request for amendment of the operating license to change the facility Technical Specifications.

(1) Implement a battery testing program in accordance with Institute of Electrical and Electronics Engineers (IEEE) Std 450-1975 (3.3.8).

(2) Implement additional Technical Specifications on reactor coolant activity (3.3.11).

(3) Add the present exclusion area boundary map (4.1).

(4) Depending on the resolution of Topic III-5.A, the licensee may develop Technical Specifications concerning the sensitivity of leak-detection systems as an alternative to other system modifications or alterations for locations where the mitigation of the consequences of a high-energy line break or leakage has been shown to be impractical (4.13).

(5) Implement inservice inspection program for concrete containment in conformance with Regulatory Guide 1.35 (4.16).

(6) Develop Technical Specifications for the component cooling water radiation monitor (4.20).

(7) Require that the overpressurization protection system be in service before the residual heat removal system is initiated (4.21.1).

(8) Develop Technical Specifications to ensure that at least one operable service water pump is aligned to each safety-related electrical bus (4.25.1).

OTHER SAFETY IMPROVEMENTS REQUIRED BY THE STAFF AS A RESULT OF SEP

The staff has determined that the following improvements are required. The licensee has either not responded to or disagrees with the position taken by the staff. The issues are discussed in the sections of this report given in parentheses.

(1) Provide adequate flood protection for safety systems from a flood of Deer Creek (4.3, 4.4, 4.5, and 4.10.2).

(2) Analyze and upgrade, if necessary, components and systems important to safety to ensure that they are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed (4.7).

(3) Perform analysis of groundwater-induced loads on safety-related structures, assuming groundwater level is at grade (4.9.1).

(4) Perform inservice inspection of water control structures (4.10.2; licensee disagrees with the part of the position pertaining to Deer Creek taken by the staff).

(5) For one penetration, install an isolation valve inside containment, and for another penetration, close upstream manual valves and leak test (4.22.2).

(6) For two penetrations, close a manual valve or make it close automatically (4.22.2).

(7) For penetrations 201, 209 (reactor compartment cooling), 308, 311, 312, 315, 316, 319, 320, and 323 (service water to and from fan coolers), change manual valves to remote manual valves (4.22.3).

(8) For penetrations 120b and 123 (gas analyzer lines), install a second automatic isolation valve on each line (4.22.3).

(9) For penetrations 201 and 209 (reactor compartment cooling), verify that this is a safety-grade closed system in containment or modify the penetrations to conform with GDC 57 (4.22.4).

(10) For each penetrations 301 and 303 (auxiliary steam heating), add one closed valve, preferably inside containment.

APPENDIX D

ACRS LETTERS TO NRC

ACRS LETTER TO NRC

CONCERNING

PALISADES PLANT SEP REVIEW

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

May 11, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SYSTEMATIC EVALUATION PROGRAM, PHASE II, AND ITS APPLICATION TO THE PALISADES PLANT

During its 265th meeting, May 6-8, 1982, the ACRS reviewed the results of the Systematic Evaluation Program, Phase II, as it has been applied to the Palisades Plant. These matters were discussed also at a subcommittee meeting in Washington, D.C. on April 15, 1982. During our review we had the benefit of discussions with representatives of the Consumers Power Company (Licensee) and the NRC Staff. We also had the benefit of the documents listed below.

The Systematic Evaluation Program (SEP) was initiated in 1977 to review the designs of older operating nuclear power plants in order to provide:

a. an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed,

b. a basis for deciding how these differences should be resolved in an integrated plant review, and

c. a documented evaluation of plant safety.

The original SEP objectives were:

1. The program should establish documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant safety issues, and should provide a rationale for acceptable departures from these criteria.

2. The program should provide the capability to make integrated and balanced decisions with respect to any required backfitting.

3. The program should be structured for early identification and resolution of any significant deficiencies.

4. The program should assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

5. The program should efficiently use available resources and minimize requirements for additional resources by NRC or industry.

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The program objectives were later interpreted to ensure that the SEP also provide safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs).

Ten plants are now included in Phase II of the SEP. The Palisades Plant is the first for which the safety reviews and the Integrated Plant Safety Assessment have been completed.

We believe that the program itself, its scope, and its methodology have been appropriate for providing the information listed in Items a. through c., above, and in meeting the objectives listed as Items 1. through 3., above. As is discussed below, the SEP can only meet objective 4. in part. With regard to objective 5., there has been a learning period. It is our understanding that the interaction between the NRC Staff and licensees is becoming more efficient.

Of the 137 topics to be addressed by the SEP, 23 were not applicable to the Palisades Plant. Twenty-four topics were found to be identical with one or more matters being reviewed by the NRC Staff in connection with the resolution of Unresolved Safety Issues (USI) or TMI Action Plan requirements. The evaluation and resolution of these topics are not included as a part of the SEP for the Palisades Plant. We believe that this was appropriate from a procedural standpoint; any other approach would have required duplication of effort within the NRC Staff or would have extended considerably the completion of Phase II of the SEP. It must be recognized, however, that because of this separation of topics, all of the SEP objectives, as listed above, have not been achieved completely at this stage of the program. For example, the documentation of objective 1 is not yet complete, the integrated and balanced decisions on backfitting did not involve all of the omitted topics (objective 2), and the assessment of safety adequacy (objective 4) is not complete.

Of the 90 topics addressed in the SEP for the Palisades Plant, 57 were found to meet current criteria or were found to be acceptable on other defined bases. In addition, as a result of modifications made by the Licensee during the review, two additional topics and parts of three others were found to meet current criteria. We have reviewed the assessments and conclusions of the NRC Staff in relation to these topics and have found them appropriate.

For all or parts of 31 SEP topics, the Palisades Plant was found not to meet current criteria. These topics were addressed by the Integrated Assessment and have been resolved in various ways: For five topics, addition or modification of equipment was required for resolution; for 12 topics, resolution required only the development or modification of procedures or Technical Specifications; and for five topics, a decision was reached that no backfit was required.

We have reviewed the treatment of these topics, and have found no reason to disagree substantially with the NRC Staff's approach, assessments, and recommended actions for resolution.

There remain nine topics for which the Integrated Assessment has not been completed, chiefly because additional information is to be provided by the Licensee. This information consists of calculations, evaluations, and various other submittals that are required by the NRC Staff as bases for its assessments and decisions. None of these topics is minor in importance to safety and most will not be easier to resolve than topics already considered. The NRC Staff expects to report the resolution of these topics in a supplemental report in the near future. Until this is done, the Integrated Assessment is incomplete by a further increment beyond that resulting from deletion of the USI and TMI topics from the SEP. As a result our endorsement and acceptance of the SEP and its application to the Palisades Plant is limited to what we have learned of the treatment of a representative group of the SEP topics. If the remaining topics are treated in a comparable manner, the objectives of the SEP will have been achieved.

The question of management performance and capability has been considered in relation to the operational history and record of regulatory compliance of the Palisades Plant. This is important because the NRC Staff has recommended changes in procedures as remedial measures for several of the SEP topics. We have noted reports of relatively recent changes in management organization, intentions, and performance. The results are encouraging but not conclusive in view of the limited length of time during which they have been observed. Nevertheless, we are satisfied with those resolutions involving procedural changes, chiefly because we are satisfied that the NRC Staff has exhibited a suitable level of concern about their effective implementation, and we are satisfied that they will continue to monitor management performance at the Palisades Plant.

A plant-specific Probabilistic Risk Assessment (PRA) was not available for the Palisades Plant. The NRC Staff utilized a limited risk assessment in portions of the Integrated Assessment, in a qualitative and subjective manner. We believe that this was done with appropriate caution and with adequate appreciation of the limitations of the analysis and the data as they applied to the Palisades Plant. We note, however, that the draft Calvert Cliffs PRA, which was utilized in the limited risk assessment, has not been available to us for use in connection with our review.

For some plants in Phase II of the SEP, and for additional plants in Phase III, it is expected that more complete plant-specific PRAs will be available. We believe that these will be useful and highly desirable as inputs to the Integrated Assessment portion of the SEP.

The Integrated Plant Safety Assessment portion of the SEP for the Palisades Plant will be documented in NUREG-0820 and its Supplements. However, the safety evaluation reports for each of the 90 topics are included only by

reference. Since these reports are an essential and important part of the SEP and constitute the only documentation of why 57 topics were found to meet current criteria or were acceptable on other defined bases, we believe that these reports should be published or otherwise made more generally available than simply by putting them in the Public Document Room.

It is expected that the results of the SEP evaluations will be among the bases used in considering the conversion of the provisional operating license for the Palisades Plant to a FTOL. We believe that these results will be very useful for this purpose. However, we defer our review of an FTOL for the Palisades Plant until such time as the remaining SEP topics have been assessed and disposed of and the topics related to the USI and TMI items have been addressed appropriately, at least in a manner similar to that being used for new operating licenses.

Our conclusions can be summarized as follows:

1. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Palisades Plant and should be achieved for the remaining plants in Phase II of the program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Palisades Plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Palisades Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI items.

Dr. William Kerr did not participate in consideration of this matter.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:
1. U.S. NRC Draft Report, "Integrated Plant Safety Assessment, Systematic Evaluation Program" - Palisades Plant, NUREG-0820 dated April 1982.
2. Letter from G. C. Lainas, Division of Licensing, USNRC, to P. G. Shewmon, Chairman, ACRS, dated 4/30/82, Subject: NRC Staff Consultants' Review of Palisades Draft Integrated Plant Safety Assessment Report transmitting Consultant Reports from R. J. Budnitz, S. H. Bush, J. M. Hendrie, H. S. Isbin, and Z. Zudans

ACRS LETTER TO NRC

CONCERNING

R. E. GINNA PLANT SEP REVIEW

ACRS-R-0987

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

August 18, 1982

Honorable Nunzio J. Palladino,
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE R. E. GINNA NUCLEAR POWER PLANT

During its 267th meeting, July 8-10, 1982, the ACRS reviewed the results of the Systematic Evaluation Program, Phase II, as it has been applied to the R. E. Ginna Nuclear Power Plant. These matters were also discussed during a Subcommittee meeting in Washington, D.C. on June 3, 1982. During our reviews, we had the benefit of discussions with representatives of the Rochester Gas and Electric Corporation (Licensee) and the NRC Staff. We also had the benefit of the documents listed below. We completed our report regarding this matter during the 268th meeting, August 12-14, 1982.

Our first review of Phase II of the Systematic Evaluation Program (SEP) was carried out in connection with its application to the Palisades Plant. Our findings from that review were addressed in a letter to you dated May 11, 1982. Our continuing review of the SEP, in relation to the Ginna Plant, has resulted in no changes in our previous findings and comments as they relate to the SEP program in general. Mr. William J. Dircks responded to some of those comments in a letter dated June 7, 1982. We find his response acceptable.

The remainder of this letter relates specifically to the SEP review of the Ginna Plant.

Of the 137 topics to be addressed in the SEP, 21 were not applicable to the Ginna Plant, and 24 were deleted from the review because they were being reviewed generically under either the Unresolved Safety Issues (USI) program or the TMI Action Plan. Of the 92 topics addressed in the Ginna Plant review, 58 were found to meet current NRC criteria or to be acceptable on another defined basis. Seven topics were later added to this category as a result of modifications made or committed to by the Licensee during the review. We have reviewed the assessments and conclusions of the NRC Staff relating to these topics and have found them appropriate.

For all or part of the remaining 27 SEP topics, the Ginna Plant was found not to meet current criteria. These topics were addressed by the Integrated Assessment and have been resolved to various degrees and in various ways.

YA01

The Integrated Assessment has not yet been completed for portions of seven topics, for which additional information must be provided by the Licensee. This information includes the results of studies, calculations, and evaluations that are required by the NRC Staff for its assessments and decisions. Six of these topics relate to structural design and the Licensee has proposed a coordinated program for their resolution. The NRC Staff has agreed to this program. The resolution of these topics will be addressed by the NRC Staff in a supplemental report that will be available for review in connection with the application for a Full-Term Operating License (FTOL) for the Ginna Plant.

For portions of ten topics included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For the remaining topics for which the assessment has been completed, the NRC Staff requires the addition or modification of structures or equipment, or the development or modification of procedures or technical specifications. Except for the three topics discussed below, the Licensee has agreed to the resolution required by the NRC Staff.

One area of disagreement relates to the groundwater level and the associated hydrostatic pressures that the structures below grade must withstand. The plant was designed assuming a groundwater elevation of 250 ft. Although limited observations from borings have shown the groundwater to be near that elevation, there has been no program of continuing measurement to demonstrate that the level does not exceed 250 ft. during periods of prolonged precipitation. In the absence of such a program, the NRC Staff has determined that the effects of groundwater should be evaluated for an assumed elevation at the surface of the ground, approximately 270 ft. for the structures of interest. We believe that such an evaluation should be made. We recommend that acceptability of the structures be based on "no loss of function" and not on arbitrary limits of stresses computed using linear-elastic assumptions.

A second topic for which resolution has not been reached relates to flooding of the site by Deer Creek, a small stream flowing into Lake Ontario in the vicinity of the plant. Flooding from Deer Creek was not considered when the plant was originally licensed; Lake Ontario was the only source of flooding considered by the Applicant and the AEC Staff at that time. Neither the NRC Staff nor the Licensee consider this question to be resolved, nor do we. Since flooding is an important matter that may have implications for other operating plants, we plan to continue our review of flood criteria, both for the Ginna Plant and on a more generic basis, and to provide our comments or recommendations when that review is completed.

The third topic for which agreement has not yet been reached concerns several containment isolation valves that do not satisfy the requirements of General Design Criterion No. 57. In view of the generally acceptable and well-considered manner in which the NRC Staff has evaluated the numerous other topics related to isolation valves, we believe that this topic should be resolved in a manner satisfactory to the NRC Staff.

As was the case for the Palisades Plant, a plant-specific Probabilistic Risk Assessment (PRA) was not available for the Ginna Plant. In its absence, the NRC Staff made careful and conservative use of a limited and essentially qualitative risk assessment, based in part on the Reactor Safety Study, for a three-loop Westinghouse plant and in part on the Interim Reliability Evaluation Program PRA for the Crystal River Plant, a two-loop Babcock & Wilcox plant. From even this limited use of a PRA, it is clear that many of the decisions involved in the SEP could be made much more rationally if plant-specific PRAs were available.

Our conclusions can be summarized as follows:

1. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Ginna Plant and should be achieved for the remaining plants in Phase II of the program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Ginna Nuclear Power Plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Ginna Plant until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI Action Plan items.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. U.S. NRC Draft Report, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant," NUREG-0821, dated May 1982.
2. NRC Staff Consultants' Review of the R. E. Ginna Nuclear Power Plant Integrated Plant Safety Assessment Report including Consultant Reports from R. J. Budnitz, S. H. Bush, J. M. Hendrie, H. S. Isbin, and Z. Zudans.
3. R. E. Ginna SEP Topic, Safety Evaluation Reports, Volumes 1 through 3, dated May, 1982.
4. U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, dated November 1980

APPENDIX E

NRC STAFF CONSULTANTS' REVIEWS OF PALISADES PLANT SEP REPORT (NUREG-0820)

REVIEW OF PALISADES PLANT SEP REPORT

BY

DR. JOSEPH HENDRIE

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Department of Nuclear Energy

(516) 282-2443
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April 27, 1982

Mr. William T. Russell, Chief
Systematic Evaluation Program Branch
Mail Stop 516
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

REF: INTEGRATED PLANT SAFETY ASSESSMENT, PALISADES PLANT, SYSTEMATIC EVALUATION PROGRAM

Dear Bill:

This letter is my technical evaluation report on the Palisades Integrated Plant Safety Assessment as set down in NUREG-0820 (the draft report). It fulfills the requirements of Task 1 of the project "Consultant Services to Review SEP Integrated Plant Safety Assessment Reports," FIN A-3367, B&R No. 20-19-20-21-1.

CONCLUSIONS

I believe the Systematic Evaluation Program, as represented by the Draft Integrated Plant Safety Assessment Report on the Palisades Plant, is fulfilling the intent of the commission when it authorized Phase II of the program in late 1977. I consider the staff recommendations for backfitting (and in other areas for no backfitting) for the Palisades Plant to be reasonable and appropriate and the bases upon which those recommendations are made to be adequate.

At this stage of the Palisades evaluation, several of the staff recommendations (requirements, really) are for further analysis, evaluation, and testing by the licensee. When the results of these efforts are in hand, decisions will have to be made about possible equipment or procedural backfitting. These decisions should be made on the same integrated assessment basis as those reported in the draft report. These "further evaluation" topics will need to be resolved before any proceedings on the full term operating license.

A number of topics that had been listed among the 137 safety topics to

be considered in the Systematic Evaluation Program reviews are currently being treated generically as Unresolved Safety Issues or as Three Mile Island Action Plan items and are, therefore, not included in the Palisades-specific review work reported in NUREG-0820. Also, there are two topics being treated generically, and thus outside the SEP Program, under other programs. One of these is the fire protection of associated circuits, being treated under implementation of Appendix R, 10CFR50, and the other is failure of main feedwater isolation, being treated under IE Bulletin 80-04. Palisades-specific resolutions for each of these topics will be needed eventually. With regard to the full term operating license, those Palisades-specific resolutions will either have to be in hand before any proceeding on the full term operating license, or the Commission will have to explicitly exclude them from such proceedings.

DISCUSSION

THE OVERALL PROGRAM

The Systematic Evaluation Program as it now functions was established late 1977, soon after I joined the Commission. Earlier work by the staff, in 1976, had resulted in Commission approval of a program to evaluate operating power reactors with respect to then-current licensing criteria, and to document the results of those evaluations and the need for any plant changes. The staff was told to prepare a list of safety topics to be considered under the program and to report back to the Commission. The staff did this in late 1977 and proposed a specific group of eleven older operating plants to be reviewed in what was called Phase II of the program. The objectives of the program were, and are, to (1) assess the safety adequacy of operating plants, (2) establish documentation to show compatibility with current requirements or justification for deviations, (3) make "integrated and balanced" decisions on backfitting, (4) give early identification and resolution of significant deficiencies, and (5) use resources efficiently and minimize impacts on staff and industry.

The need for some sort of safety review of the older plants in particular had been obvious for some time. The ACRS had been recommending a systematic review of operating plants for many years. There were always questions arising, particularly with Congressional staff and committees, about whether the older plants, designed and constructed to an earlier set of safety standards, still met the Commission's current regulations. Along with the need for some assessment of the safety adequacy of the older operating plants, it was also clear that

it would be most useful to have current documentation which would show the compatibility of the design of those plants with current criteria or the basis for permitting deviations to exsist.

The assessement of safety adequacy by the staff for the Palisades plant, as evidenced in the draft report NUREG-0820, is in my view, a competent and sufficient job on the items covered thus far. There remain, of course, the Unresolved Safety Issue topics and the Three Mile Island Action Plan topics for Palisades, as well as the two other generic items being pursued outside the SEP. Assuming that these will be treated for Palisades in the same fashion as the topics that have been reviewed and reported upon thus far, the safety assessment work has been a thorough and careful job that meets the Commission's intent in this area.

Not every item conceivably related to safety at a nuclear plant is encompassed in the SEP, of course. The original culling of more then 800 possible items for consideration down to the final Phase I list of 137 topics indicates that very clearly. But, in my view, all of the important safety matters are being covered under the SEP for these older plants, and that is what the Commission wanted.

The documentation of the assessment, which is mainly in the safety evaluation report letters, one for each of the SEP topics dealt with in the Palisades review, seems to me to be sufficient for the purpose. NUREG-0820 summarizes the assessment, deals with each of the 31 safety topics on which there were deviations and for which questions of backfitting arose, and includes in its appendices the Sandia report on probabilistic risk assessments of some topics and an Oak Ridge report on the operating history at Palisades. I presume that a supplement or supplements to NUREG-0820 will be issued to cover the outcome of the Unresolved Safety Issue and Three Mile Island Action Plan items and also the results of current analyses and evaluations being carried out by the licensee.

A major element in my own approval of the Systematic Evaluation Program in 1977 was the proposition that these plant reviews would be done on a integrated and balanced basis in recognition of the fact that they were dealing with plants that had been operating more or less successfully for some time. I would not have agreed to an SEP in which the review was to be done as if it were a new license, item by item, with all of the i's dotted and t's crossed. What was needed in my view, if the work was to be done at all, was an overall safety assessment of the plant as an entity, looking for places where safety upgrading was clearly needed. After reviewing the Palisades documents, I

conclude that the SEP staff has done a good job in performing t..t sort of "integrated and balanced" assessment.

Of the other two program objectives, early identification of significant deficiencies and efficient utilization of resources for both staff and industry, I note that no urgent safety deficiencies were found at Palisades, so there was no need to exercise that objective. As for the last objective, I am inclined to think the utilization of resources has been done reasonably efficiently, although the job has taken a lot longer then originally projected. Three Mile Island bears a substantial responsibility for this, of course.

I conclude that the staff, in carrying out the Systematic Evaluation Program assessment of the Palisades Plant, has fulfilled the Commission's intent as reflected in the major program objectives laid down in the staff papers that are the basis for the SEP.

POL-FTL CONVERSION

One of the Commission's aims in establishing the SEP was that the safety assessment work and its documentation would serve as a primary basis for the conversion of provisional operating licenses held by five of the plants in the Phase II program to full term operating licenses. I recall, in fact, that this was a major consideration for me in approving the program. Those provisional operating licenses, automatically renewed every 18 months, had long been an embarrassemnt. Conversion to full term operating licenses was going to be necessary at some point, and the sooner the better. The SEP effort offered precisely the kind of safety review that was needed for the conversion, that is, one which took an integrated view of the whole plant and its operations from a safety standpoint.

The material at hand from the Palisades SEP review will be the primary documentation of the staff work and the plant status in going forward with conversion of the Palisades Provisional Operation License. The material developed thus far will serve the purpose, I think. This objective of the SEP, then, is also being achieved. There are various parts of the review that are still to come, of course. The results of licensee evaluations, analyses, and tests now being done will have to be considered and any possible backfitting matters settled and documented in a supplement to NUREG-0820. There are two matters being treated generically under other programs that are, nevertheless, listed among the Palisades SEP topics. These are the fire protection of associated circuits and main feed Water isolation. I presume that both of these will be resolved for Palisades

specifically and the results of those resolutions included in the Palisades POL-FTL conversion documents. Then there are the major outstanding items, the Unresolved Safety Issue topics and the Three Mile Island Action Plan topics that are being treated generically under those two programs and outside the SEP. These include important safety topics for Palisades.

The POL-FTL conversion proceedings for Palisades cannot go forward until the results of the USI and TMI resolutions for Palisades are documented, unless the Commission specifically removes these matters from consideration in the POL-FTL conversion proceedings. This latter course is a possible one and would be justifiable on the basis that when the generic resolutions of the USI and TMI topics are achieved, the operating license (possibly a full-term license by that time) for Palisades would be amended to include those resolutions.

Although it is a possible course and could be justified as noted, I am inclined against it if there is any hope of achieving Palisades-specific resolutions of the outstanding USI and TMI topics. The reason is that setting them aside for later treatment as license amendments exposes the process to a second possible hearing when the USI and TMI amendments to the license are imposed. I expect that on most occasions these days when the opportunity for a hearing is offered, there will be a hearing. So, it would be handy all around if the USI and TMI outstanding topics could be resolved for Palisades on a schedule that would allow their inclusion in the proceeding on the POL-FTL conversion.

THE STAFF SAFETY REVIEW

The Palisades Plant was reviewed against the 137 SEP safety topics. These are listed in Appendix A of the draft report. These 137 topics were sorted out in early 1977, following Commission approval of the initial SEP proposal and in preparation for the October 1977 paper to the Commission. In spite of their age, it strikes me that the 137 safety topics still form an appropriate list of areas for review of these older plants.

Of the 137 topics, 23 are not applicable to Palisades and were deleted from the review. In addition, 24 topics were deleted from the Palisades review because they are being covered generically under the Unresolved Safety Issues or the Three Mile Island Action Plan programs. The remaining 90 topics are reported upon in the draft report NUREG-0820.

The 137 SEP topics are heavily oriented toward design matters. Only

three are specific to the operation of the plant per se. These are XIII-1, Conduct of Operations; XIII-2, Safeguards/Industrial Security; and XVII, Operational Quality Assurance Program. The first, Conduct of Operations, is a TMI issue and is not treated in the draft report. (There is, however, a report commissioned by the SEP from Oak Ridge National Laboratory on the Palisades operations: This is given in Appendix F of the draft report.) The other two topics are covered in the Palisades review and both were found to be satisfactory.

The 90 topic reviews came out in one of three ways: (1) Palisades either is consistent with, or equivalent to current licensing criteria. 57 of the topic reviews came out this way. (2) Palisades is not consistent with current licensing criteria, but the licensee has implemented or committed to implement equipment or procedural changes that make it consistent with or equivalent to current criteria. Two topics came out this way. (3) Palisades is not consistent with current licensing criteria and the topic was turned over to a staff team for a integrated assessment and possible backfitting recommendations. 31 of the topics fell into this category. No urgent safety problems were identified of a nature that required immediate action. Current licensing criteria are taken from the current Standard Review Plan (July, 1981).

In 14 of the 31 topics for which backfitting was a possibility, a probabilistic risk assessment was found to be possible either on the whole topic or on some subsection of it. The risk assessment was done on a relative basis by Sandia Laboratories. Sandia compared the Palisades as-is system with a backfitted system to obtain a measure of the reduction in risk (primarily in the probability of occurrence) that might follow from backfitting. The Sandia report is included in NUREG-0820 as Appendix D. Since there is no complete probabilistic risk assessment for Palisades, or even for a Combustion Engineering plant, the Sandia work had to depend on an unpublished risk assessment for Calvert Cliffs as a baseline. The resulting assessments of safety importance and of benefit in risk reduction from backfitting are necessarily rough but are still useful inputs to be considered in the overall assessment of the topic.

The results of the integrated assessment of the 31 safety topics in which Palisades had significant deviations from current licensing criteria may be tallied as follows. The 31 topics include a number of topics which have several sections that had to be treated essentially as separate reviews. If one counts all of these separable issues, the 31 topics become 58 subtopics

or issues. Of the 58 issues considered:

- 2 are being treated generically outside the SEP,
- 23 were found to require no backfitting measures, and
- 33 were found to require some backfitting measures.

Of the 33 issues that were found to require some backfitting measures:

- 7 required equipment changes or additions,
- 14 required procedural changes or additions, and
- 12 required further analysis, evaluation, or testing, which could lead in turn to requirements for equipment or procedural changes or additions.

Of the 21 equipment or procedural changes and additions, 12 led to new Technical Specifications being required.

In addition, during the review, the licensee made or committed to various equipment changes and modifications under 5 topics. These would have added to the 31 topics or the 58 issues if they had not been fixed during the review.

So, the integrated assessment team has, thus far, required equipment changes or additions in only 7 out of 58 safety issues before it. In addition, in 14 cases, issues were settled by procedural changes or additions. Nevertheless, it seems clear that the integrated assessment team has not come down blindly for backfitting no matter what the cost or safety benefit.

The October, 1977 staff paper, which was the basis of Commission approval of Phase II of the SEP, noted that when deviations from current licensing criteria were identified there were a number of alteratives or combinations of the same that would be considered as a basis for acceptability. These included acceptance of the deviation as not significantly decreasing the safety level, use of non-safety grade systems to perform safety functions, administrative or procedural changes to enhance safety system reliability, augmented surveillance programs for the same purpose, and selected backfitting. Deviations from current criteria were to be acceptable if the staff evaluation showed that the plant would respond satisfactorily to the various design basis events and the probability of those or the consequences were not significantly higher than for plants licensed in accordance with current criteria.

In reviewing the safety evaluations of the 31 topics where significant deviations have been identified, I conclude that the SEP staff has followed that direction faithfully. They have looked carefully at the risk reduction and safety benefit to be achieved by any changes and have utilized all of

the available alternatives in arriving at these final judgements. I think the staff bases for requiring equipment changes or modifications in the few cases where that has been done, for requiring procedural changes in other areas, and for concluding that no backfitting is required in yet other areas are adequate and reasonable and are consistent with the Commission's directives of long ago. I am particularly pleased to see the staff willing to declare that there is no need for backfitting in those cases where it offers little or no reduction in risk and would have substantial impact on the plant if required. That has not always been a characteristic of staff reviews.

It is going on 5 years since the SEP Phase II came before the Commission for approval. I voted for it with a certain amount of trepidation. I had some concern then over the staff's ability to do a balanced assessment on an older operating plant and to come up with results that were meaningful from a safety standpoint and did not simply end up requiring total conformance with current criteria regardless of the safety benefits. That concern did not abate much in the years following and I used to confront the bright-eyed proposers of a SEP Phase III with the direction to go back and produce something from Phase II and then we would see. Now we have the first product from Phase II. I think it is a good job. My compliments to the staff.

Sincerely,

Joseph M. Hendrie

REVIEW OF PALISADES PLANT SEP REPORT

BY

DR. STEPHEN BUSH

PALISADES PLANT

A CRITIQUE OF THE INTEGRATED PLANT SAFETY ASSESSMENT SYSTEMATIC EVALUATION PROGRAM

S. H. Bush

Since Palisades is the first plant reviewed under the Systematic Evaluation Program, the approach taken and the criteria used to establish the acceptability of assessment are somewhat tentative, particularly because there has been no opportunity to interface with authors and other reviewers. Two suggested benchmarks are:

- Does the report meet the original AEC/NRC Commission Charter for SEPs.
- Are the items identified as problems adequately described, including justification of their resolution.

An examination of documents SECY-76-545 and SECY-77-561 provided some insight into the approach used to handle SEP plants. The five program objectives can be used as criteria for measuring compliance. The suggested approach for handling deviations can permit an assessment of the resolutions suggested in the Palisades report. These criteria follow.

The following five objectives of the program were established by the Task Force:

1. The review program must assess the adequacy of the design and operation of all currently licensed nuclear power plants.
2. The program should establish documentation which shows how each operating plant compares with current criteria on significant safety iss s, and provide a rationale for acceptable departures from these criteria.

3. The program should provide for the capability to make integrated and balanced decisions with respect to any required backfitting.
4. The program should be structured for early identification and resolution of significant deficiencies.
5. The program should efficiently utilize available resources and minimize requirements for additional resources by NRC or industry.

The planned systematic evaluation would establish the adequacy of all operating power reactors with respect to safety and provide clear written documentation bases for this conclusion.

When deviations from current licensing criteria are identified, the following alternatives (or combinations of alternatives) will be considered as a basis for establishing acceptability:

1. The deviation can be justified as not significantly decreasing the level of safety.
2. Use of non-safety systems to perform safety functions.
3. Administrative or procedural changes to enhance system reliability.
4. Augmented surveillance programs.
5. Selected backfitting to enhance system reliability.

Presumably one critical evaluation of Appendix A will be sufficient on the assumption that these items will remain unchanged in the future. While Appendix B covering generic issues may change somewhat, one review as to adequacy should be sufficient. Obviously, Appendix C will change because of plant and site specificity. Appendices E and F will need review on a case-by-case basis.

Examination of Appendices A, B, and C unearthed some problems. The wording, references and approach used with the items in Appendix A reveal the "mind set" of the 1976-77 period. Personally, I feel that some of the strong positions taken then have weakened in the past 4-5 years. An example might be valve lockout. As predicted some of the locked out valves have been found to be in the wrong position so the effects of an accident would be exaggerated.

I suspect a probabilistic approach could lead to dropping others; however, the option appears to exist in the so-called "lesser safety significance" approach.

With regard to Appendix B as related to A, I am at a loss as to why some of the unresolved safety issues were ignored. Specifically, issues A-11, A-31, A-45 and A-49 were not cited. If these were included, some other items would shift to the generic packet. While I understand the words regarding folding in the USI and TMI issues, it is not immediately obvious how this will be accomplished.

I suspect that the issues in Appendix A, if written in 1982-82, would differ substantially from the words generated in 1977; however, those words can be accepted.

SECTION 1

An item of major concern becomes apparent in the listings on page 1-7 and in Appendix F. While the number of LERs arising from personnel or procedural errors is not large, the safety significance of some of the events is substantial, particularly with regard to loss of containment integrity and improper positioning of safety-related valves. These events extend over a sufficiently long period that is indicative of an indifference on the part of top management to take appropriate action. In my opinion the document does not stress this area sufficiently. Unless there is positive evidence of an improvement in operator actions, I question approving a full-term operating license.

SECTION 2

Explanatory only--no comments.

SECTION 3

The positive actions taken to resolve issues III-6, VII-3, VIII-2, VIII-3B and VI-6 are considered appropriate. My personal opinion is that some of the changes under III-6 may not have contributed much to plant safety.

SECTION 4

In essence, this section represents the actions and bases for the actions taken including a factoring in of the PRA in Appendix B.

II-1-A no comment; no problem.

II-38, 81.C Pending; probable backfit.

III-1 Positive actions that should provide missing information and enable decision as to acceptability of various items.

III-2 A good example of accepting alternate approaches when deviation occurs. Instead of backfitting, it is recognized that sources of water can be made available. Emphasis is on clearly defined procedures covering use of alternate water sources than on upgrading or backfitting.

III-3-C The positions of staff and utility are apparent. I would have thought this to be an economic problem that would become apparent during operation rather than under accident conditions. I agree with staff.

III-4-A I applaud the decision not to backfit. It's appropriate.

III-5-A, III-6 I disagree on philosophic grounds with this item. In ten years of review I have yet to find a case where piping failed from seismic loads and no breaks result from an unrealistic application of the design load cycles. Current analytic technique yield a false picture of piping response that seemingly is not recognized.

III-7-A No disagreement--okay.

III-7-B Primarily a bookkeeping activity to provide analytic answers.

III-7-C I understand the need to do another examination for delamination. I do not understand an arbitrary five-year repeat. We don't require that on embedded flaws in vessels.

III-8-A May shift to generic.

V-5 A realistic approach. I agree with staff analyses.

V-10-8 Action taken resolves issue.

V-11-A This had potential to overpressurize and fail piping. The action only resolves it partially since case of released flapper is not covered.

VI-2-D, VI-3 I agree with decision and PRA value. No action required.

VI-4 Removal of threaded piping is appropriate. Other decisions acceptable.

VI-6 Forced action taken--no issue.

VI-10-A No action.

VII-1-A A good example of use of PRA to require revision or accept status quo.

VII-3 DC power obviously is important. Basically handled as generic problem. Other actions based on a realistic assessment of tradeoffs.

VIII-3-A Important issue. Must assume loss of diesel generator plus offsite power.

VIII-4 Action taken.

IX-3 Presumably fix will be procedural in nature. Not clear. Second item procedural plus modification.

IX-5 Analytic only--not complete.

IX-6 In essence a generic backfit item.

XV-2 I am not surprised regarding the uncertainty in failure rates. Basically, this will be handled generically.

XV-12 A realistic approach to the problem.

With regard to equipment and design items, the authors addressed to a major degree the SEP task force objectives as well as applying the tiered criteria to resolve deviations. Generally, the approach is even-handed, not requiring backfit arbitrarily. I am less satisfied with the handling of operating history.

Appendix F points out the high incidence of loss of power. This combined with some of the operator errors listed could yield a definite degradation in safety margins.

REVIEW OF PALISADES PLANT SEP REPORT

BY

DR. ROBERT BUDNITZ

)

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2000 Center Street Berkeley, California 94704 415/526-5111
Room 418

RECEIVED
APR 19 1982

15 April 1982

Mr. William T. Russell
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear ~~Mr. Russell~~ Bill:

This letter comprises my report to you under Purchase Order # DR-82-0961, in which my assignment has been to review Draft Report NUREG-0820, "Integrated Plant Safety Assessment: Systematic Evaluation Program, Palisades Plant". As you know, I received a copy of this report personally on April 2nd, when I was visiting Bethesda on other business. I arrived back here in Berkeley on April 7th, and after verbal authorization from Ms. Arlene McNulty of NRC Division of Contracts on April 9th, I began the review process in earnest. Unfortunately, I am departing on April 16th (tomorrow) for a two-week business trip, so my review has had to be squeezed into the few days between April 9th and 15th, plus the time I put into it after I got the draft report on April 2nd.

Regarding the mission of the SEP, I have used as primary references a pair of Commission papers (SECY-76-545 and SECY-77-561) that you furnished. I understand that these together comprise the 'charter' for the SEP effort. Of course, during my two years at NRC (1978-80) I learned a lot about the SEP and therefore have considerable additional background as to its goals, methodology, and constraints.

While I have nothing in writing telling me my own scope of work, I have read the scopes of work for two other reviewers (Drs. Bush and Hendrie), and I have assumed that my own scope is identical. I understand that the objective is "to provide an evaluation of the adequacy of the rationale used by the staff in identifying and making recommendations for backfit requirements." I have interpreted this charter slightly more broadly, and my comments will reflect my broader interpretation. To be specific, I will provide discussion touching on each of the following questions:

1) Is the Palisades SEP report asking and answering the right questions ?

2) What implicit policy-type decisions do I detect in the report ? Is their rationale appropriate ?

3) Is the review methodology appropriate ?

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4) Are important issues left out of the review ?

5) What has been included in the review that might have been omitted without significant compromise ?

6) How adequate is the rationale used to identify and recommend backfit requirements ? (This question is the specific objective of the review.)

Because I have been short on time, I have not provided herein any comments on specific safety topics. I do have several specific comments, with varying degrees of importance, which I will assemble into a coherent package during my trip over the next two weeks. If it seems useful later, I can provide additional material to you upon my return, after May 2nd.

A. Is the Palisades SEP Draft Report asking and answering the right questions ?

In a narrow sense, I believe that the answer to this question is affirmative: that is, the original charter seems to emphasize reviewing the older plants against modern review criteria (the modern Standard Review Plan, modern regulatory guides and standards, etc.), with the subsidiary goal that for plants with Provisional Operating Licenses the SEP review would form part of the basis for conversion to Full Term Operating Licenses. To the extent that these objectives have guided the SEP effort, they have been quite successfully fulfilled, in my view. I find that there has been a systematic analysis of the areas where the Palisades plant review, were it being conducted by the NRC staff today, would have been different: in some areas the review procedure would have been different, and in others the plant or its operating procedures would have been different.

When I used the word 'narrow' in the first sentence of the paragraph above, I meant it in its purest form: that is, I realize that it is extremely important that every item on the 'list' be discussed properly, and its resolution documented. I find that the draft report has accomplished this effectively. As one who has generally been uncomfortable with the (apparently ubiquitous) need to get papers into the file covering every gnat's eyebrow, I find the detail contained in some of the explanations and resolutions to be a little extreme .... but I do recognize the legitimate reason for this, namely that bringing Palisades into line and up-to-date with the large number of newer operating units is important in its own right.

My discomfort arises from the following perception: I personally believe, and have believed for some time, that plants such as Palisades have been built and operated in a manner that assures "adequate protection of the public health and safety", in the sense that the NRC Commissioners and staff have used that phrase or its generalized counterparts like 'no undue risk'. However, this belief does not rest upon specific, well-founded grounds except for the strongly affirmative safety record of the industry to date; therefore, analyses that tend to confirm it are always important. In this regard, one philosophical rationale for undertaking the Systematic Evaluation Program has been to look at

older plants like Palisades, attempting to uncover any safety concerns that might cast doubt on the "adequate protection" determination. In this regard I have reached two conclusions after reading the draft SEP report. The first is that none of the safety issues treated seem in my view to have turned out to be highly important to safety, after analysis .... and this conclusion comforts me a good deal: I can almost hear myself breathing more easily. Second, and in some ways more significant, is my conclusion that a few quite important safety issues are absent from the analysis in this draft report ! I elaborate as follows: if somebody asked me for my personal opinion about what safety issues might compromise the judgment that Palisades poses no undue risk, I would list several items that are broadly encompassed by the USI (Unresolved Safety Issues) and TMI (Three Mile Island Action Plan) categories. Thus, I continue to be worried about things like systems interactions (USI A-17), station blackout (USI A-44), control systems issues (USI A-47), the full range of human factors concerns, and the dependency of safety systems on crucial support functions like instrument air, service water, and electrical distribution buses of uncertain reliability.

The fact that NRC is systematically addressing these USI and TMI issues gives me comfort. In my view it is very likely that all of them will be resolved sooner or later, that all of our plants will somehow be safer because of it, and that the safety imporvements will be highly cost-effective. Nevertheless, I believe that the draft report I have in front of me is somehow inadequate or insufficient to the extent that it does not highlight this key point. I would feel better if the report had something like the following, up front somewhere, to guide the reader:

> "The regulatory staff recognizes that several of the most important safety issues have not been addressed or resolved in the course of this SEP effort, in each case because they are being coped with through other regulatory efforts: in particular, the Unresolved Safety Issues list and the Three Mile Island Action Plan list contain some issues whose safety significance is probably far greater than a majority of the issues dealt with and resolved herein."

In summary, my answer to the question posed above ("Is the report asking and answering the right questions ?") is that while it asks most of the right questions, it finds itself unable to answer a reasonable fraction of them.

Phrasing my concern another way, I find a lot of what is in this draft report to be operating in the strange make-believe land of the traditional NRC approach to regulation, an approach where regulation per se survives as important separate from safety. One concrete example of this is the (luckily

few) places where, instead of assuring by other means that the licensee carries out a certain procedure, the staff wants a change to the Technical Specifications. I had the impression that the staff was moving toward a philosophy of having less specificity in tech specs .... and if it isn't so moving, I believe it ought to ! Yet here is the SEP effort sticking more little stuff into tech specs. Isn't there some other, better way ?

B. What implicit policy-type decisions do I detect in the report ? Is their rationale appropriate ?

I detect several policy-type decisions that I agree with. Perhaps the most important is the general feeling that I get in reading the report that Palisades is, indeed, 'adequately safe'. This feeling pervades the text of the report as I read it, and it apparently has played a part is some of the decisions on whether urgency is required for various backfits.

Another important policy decision is the strong presence of the concepts of PRA (probabilistic risk assessment) as a valuable tool in safety decision-making. I endorse this with delight: I feel that the way PRA has been used is just about right. It has been used for its insights into relative safety importance, but for not much in the way of quantitative information. (Of course, this is partly because there has been no PRA done on Palisades itself; the Sandia-written appendix only references a comparison between systems at Palisades and at a mysteriously-unnamed different plant of CE design.) The Sandia write-up on the PRA analysis is lucid, and explicitly recognizes the major uncertainties in any quantitative conclusions.

Another key policy-level decision seems to have been that hardware fixes should be required only if no other type of backfit or procedural arrangement is available. I applaud this decision. Conscious efforts to avoid unnecessary backfits are, in my view, an important element in NRC's regaining credibility with the licensees.

C. Is the review methodology appropriate ?

I am pleased to report my finding that the methodology used in the report is appropriate and adequate for the purpose. As mentioned above, I am especially pleased to note that PRA methods have been used to rank the safety significance of several of the issues, and that PRA insights have assisted the staff in deciding on the importance or urgency of required changes. (I could quote the Lewis Report here, but I will restrain myself.)

The methodology gives different depth of treatment to issues of differing safety significance, and this is fully appropriate. I especially applaud the concept of an integrated assessment in which a large number of issues are viewed in sum rather than one-by-one. This integration affords the analyst broad insights into urgency and cost-effectiveness; and affords the reviewer or critic the chance to grasp the whole SEP analysis more fully. I think that whoever has brought about this conceptualization of the SEP effort should be congratulated for clarity of thought.

I also find the approach used by the Oak Ridge group in analyzing operating experience at Palisades to be a good one. Their logic in identifying those events with real safety significance seems to be fully satisfactory, and there are some excellent discussions of specific topics (especially about control rod drive mechanism problems and partial/full loss of offsite and onsite power.)

The one part of the methodology that leaves me a little uncomfortable is the linkage of the analysis, at least in a structural sense, to the outmoded issues list compiled in about 1977. The list itself (the definitions in Appendix A of the report, for example) contains some examples of thinking about safety/regulation/retrofits/NRC-licensee interactions that are today outdated, or at least overtaken by the events following Three Mile Island. However, the implementation of the methodology overcomes much of the difficulty imposed by the use of the outmoded list and definitions: there are several examples of more up-to-date thinking about issues.

D. Are important issues left out of the review ?

I have already discussed my discomfort that several important issues are not analyzed in this draft report because they are being coped with through a different regulatory mechanism (USI, TMI, etc.). I understand the rationale for this, and accept it prima facie.

I also have discomfort about the omission of a collection of issues involving management. Specifically, I know that various utility managements are viewed in different ways within the NRC staff: some are thought to be more competent than others, without necessarily implying that any one or more of them are insufficiently competent. What struck me as I read this draft report is that I cannot, for the life of me, figure out from it how Consumers Power's management is viewed ! (The discussion on page 1-6, penultimate paragraph, is the only clue I found as to what NRC thinks about Palisades management.) For all I know, they are thought to be 'the best utility around', or 'the worst', or whatever. Since everybody now appreciates how crucial good management is to safety, some specific treatment of this issue would seem to be called for.

The same comment applies to utility in-house engineering competence. Some utilities have very fine engineering staffs, while others are weaker, relying instead on outside assistance. Again, I cannot figure out where Consumers Power fits into this spectrum, yet engineering competence, like management competence, looms large as a key element in safety.

Finally, the entire SEP seems to give insufficient treatment to the human factors and control systems side of safety. Even considering that many human-factors and control systems issues are bound up in the TMI Action Plan list, I would have felt better had there been more discussion of them, especially in the integrated assessment part of the report.

E. What has been included in the review that might have been omitted without significant compromise ?

In one phrase, "not much" .... except for the apparently ubiquitous need to cross all the t's and dot all the i's.

F. How adequate is the rationale used to identify and recommend backfit requirements ?

This is the question that was asked directly of me as a reviewer.

I already mentioned, and will repeat here, my finding that the rationale for decision-making is fully satisfactory. I find the staff's thoroughness, issue by issue, to be commendable. I find that the use of PRA as an aid to engineering insight is at just about the right level. I am pleased with the apparent decision not to seek hardware changes except in those few areas of high safety significance where no other remedy could be identified.

Summary

I will summarize by stating that I believe this first SEP report has been quite successful: the metaphor of the laundry list that has been cleaned up is appropriate. Maybe Palisades can get a regular operating license now, for one thing; and maybe the utility staff and the regulatory staff can go on from this mop-up activity to think hard about the real issues of safe operation of Palisades, issues hardly dealt with in the analyses within this draft report.

Finally, I do think it is important to state my view that it is only in retrospect, after the analysis, that one is at liberty to characterize the SEP list for Palisades as a 'laundry list': beforehand, we didn't really know what would crop up. So in that regard the activity has been successful indeed.

Robert Budnitz 15 April 1982
Robert J. Budnitz

REVIEW OF PALISADES PLANT SEP REVIEW

BY

DR. HERBERT ISBIN

W Russell

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April 23, 1982

To: Project Officer W. Russell
Mailstop 516
7920 Norfolk Avenue
Bethesda, Maryland 20014

From: H. S. Isbin H. S. Isbin

Draft
Review of Integrated Plant Safety Assessment
Systematic Evaluation Program
Palisades Plant
Draft NUREG-0820

Enclosed please find my draft review in response to your request to "...provide an evaluation of the adequacy of the rationale used by the staff in identifying and making recommendations for backfit requirements." In addition to Draft NUREG-0820, I received SECY 77-561 (October 26, 1977), SECY 76-545 (November 12, 1976), a November 15, 1977 memorandum from S. J. Chilk to L. V. Gossick, and a draft Statement of Work. I have not had the benefit of any discussions with the SEP staff nor with any reviewers. Please let me know if you desire any changes in the focus of my review.

The highlights of my review are as follows:

- The planning used for SEP is outstanding from the point of view of identifying safety items.

- The objectives have been well conceived; however one major objective may have been inadvertently omitted in the NUREG report.

- The review of operating experiences needs to be updated and augmented.

- Limited assistance was provided by the probabilistic risk assessment for this plant.

- The reporting of the Topics and the ensuing approach to the decision making, in general, are well done.

- Too many events and changes have occurred in the past three years to be able to evaluate whether the SEP program is efficiently and economically using NRC and Industry resources.

- An important finding is that no SEP Topic was considered to be of sufficient importance to require a prompt resolution.

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- Attention has been focussed on achieving an "...integrated and balanced..." decision, considering that the SEP program is being carried out in conjunction with major NRC and Industry efforts for implementing TMI Action Plan Items, and responding to IEBulletins and Generic Letters. Resolution of Unresolved Safety Issues remains a continuing activity along with mandated annual reports to the Congress concerning identification of any new issue: Overall assessment of safety of the plant must utilize all these inputs.

Additional SEP supplements have been planned for the Palisades plants. T! Status of the SEP Topics, presented in Appendix A, is for the date April with some having been updated to May 1981 for inclusion of TMI tasks, USI several IE Bulletins. I understand that no changes were made in Topic definitions. Has consideration been given to updating all the SEP Topics regarding status and References?

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DRAFT REVIEW

by

H. S. Isbin

COMMENTS ON SECTIONS OF NUREG-0820

1.2 Systematic Evaluation Program Objectives

Only three objectives are presented. I believe that the fourth objective is essential and from the referenced material I extract

> "and (4) an overall evaluation of all safety topics evaluated in the SEP and other ongoing programs..."

I would emphasize overall, all, other ongoing programs.

I assume that the presentation of the original five SEP objectives is to augment the present objectives. Any program that seeks "...to make integrated and balanced decisions with respect to any required backfitting" and to "...efficiently use available resources and minimize requirements for additional resources by NRC or industry "merits our standing ovation.

A variety of actions taken during the last three years on generic matters, including the TMI related items, have considerably altered priorities. In my review, I have chosen to focus on what elements need to be included to achieve the "...integrated and balanced..." backfitting decisions.

1.4 Summary of Operating History and Experience

This section is inadequate because it is not updated.

The ORNL detailed review is a worthy study up to and including the year 1979 and represents an "external" appraisal. More emphasis needs to be given on implementation of corrective actions.

The summary of Escalated Enforcement Actions presents significant events through March 1981. I suggest including more "internal" reviews, including the Region III "Systematic Assessment of Licensee Performance" (SALP), and the periodic Inspection Reports by the resident inspector(s) and by the special teams. The updating might include the Licensee's Annual Report of Changes, Tests and Experiments.

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The listing of the civil penalties and orders is not as important as the thorough evaluation of the Licensee's response in terms of corporate policies, management and staff control measures, training and requalification programs, procedures, and quality assurance. What are the lessons learned and how are the improvements being implemented? Are these actions contributing to overall plant safety?

Corrective features are embodied in the TMI Action Plan. Analysis and feedback of operating experiences must be achieved in a realistic program.

The SEP report does not reflect any improvements in training nor even the considerable augmentation of staff (which I assume must be taking place).

In my opinion, it is too early for the NRC to reference INPO reports and documents; however, if the Licensee has made or is making improvements as a consequence of INPO evaluations, such information assists in making "...balanced and integrated..." decisions.

Programs involving SEE-IN and NOTEPAD should be checked for updating operating experiences and reference should be made to the NRC Generic Letter 82-04. Have the SEP reviewers made any use of the improvements underway on handling and managing the collation of LERs? Is the Sequence Coding and Search System operational?

A feature of the SEP program that I had expected to find, but did not, is concerned with "aging" of components and systems. Have there been any discernible trends? Not all events are reportable, and thus some important trends might be missed if recourse is made to just LERs. The cooperation of the maintenance and inspection groups is needed. Are the IE Information Notices helpful in ascertaining whether any special aging effects could have an impact on safety?

2. Review Method

2.2 Selection of Topic List

The identification of the more than 800 candidate items took place in 1976 and 1977, and the methods used are impressive. The process of reducing this number to 90 topics applicable for the Palisades SEP review has an acceptable rationale, providing all current items involving Unresolved Safety Issues, TMI Action Plan Items, and other generic matters are to be included. The draft SEP report indicates that a supplement is to be issued which will designate the status of the USIs and TMI Action Plan Items. I consider this supplement to be a key factor in balancing overall decisions to be made on the SEP items. The magnitude of this task should not be underestimated.

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2.3 Topic Evaluation Procedures

The two methods used for the preparation of the final Safety Evaluation Reports for the 90 topics have involved Licensee participation to ascertain that correct information was used. I have not seen any of the SERs, but I think that the approach used is good.

The finding by the NRC Staff that no topic identified in the SEP review required immediate action is significant.

Topics were grouped into categories regarding no further action, action initiated by the Licensee which is acceptable to the NRC, and finally those which require decisions on whether backfitting is needed. This approach is logical.

2.4 Integrated Plant Safety Assessment

The overall planning of the SEP review to achieve a "...balanced and integrated..." decision on each topic appears to be good. The approach used by the NRC Staff appears to be consistent with the designated objectives.

I was not able to judge how the SEP work loads have impacted on the Licensee's resources. Nor have I been able to judge how the NRC plans to mold the backfitting decisions with decisions made and to be made on the USIs and TMI Related Safety Items. Perhaps this subject will be treated in a supplement.

COMMENTS ON SELECTED TOPICS

3.3 Topics for Which Plant Design Meets Current Criteria

Based on Modifications Implemented by the Licensee

3.3.1 Topic III-6 Seismic Design Considerations

The Licensee must be responding to IE Bulletins, dealing for example with structural integrity of masonry walls and safety related piping systems. The report would be improved if the relationships of the programs involved for the IE Bulletins to the SEP concerns were clearly presented.

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Integrated Assessment

4.6 Topic III-2 Wind and Tornado Loadings

4.6.1 Safety Injection and Refueling Water (SIRW) and Condensate Storage Tanks

The approach taken by the Staff is good.

"...if the SIRW tank or condensate storage tank is lost..."

Rather than just "or", don't you mean either or both? Further, wouldn't you need to comment that the failure(s), in themselves, do not produce any undue flooding effects?

This is one of the few items where the TMI Action Plan, Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents," is mentioned. This SEP topic is a part of a much broader and more complete task, with priority on Licensee's resources to be given to the TMI Item.

4.10 Topic III-6 Seismic Design Considerations

Once again, for an "...integrated and balanced..." approach, SEP concerns need to be factored into the broader areas being addressed through responses to IE Bulletins.

4.15 Topic V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

The rationale used by the Staff in arriving at conclusions fits the objective set forth for SEP. Is there any impact on radiation exposure to workers?

4.16.2 Use of Safety-Grade Systems for Safe Shutdown

On page A-42, modification of the Branch Technical Position RSB5-1 is being suggested. What is the current status?

4.19 Topic VI-3 Containment Pressure and Heat Removal Capability

The emphasis of this resolution is on the capability of the containment to withstand the increased pressure resulting from a two-steam generator blowdown. A "...balanced and integrated..." approach should avoid exacerbating other possible issues. For example, would there by any impact on resolving the concerns presented in IE Bulletin 79-01B, dealing with the environmental conditions for the qualification of safety-related electrical equipment?

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4.23 Topic VII-1.A. Isolation of Reactor Protection System from Nonsafety Systems, Including Qualification of Isolation Devices.

No reference is given to the completion of Technical assignment Control No. 6696, nor whether there have been any continuing studies.

4.24 Topic VII-3, Systems Required for Safe Shutdown

I had expected to find a discussion on what can be done outside the control room "...to achieve and maintain a safe shutdown condition of the plant..." (See A-66).

The SERs started with the general topic and then determined specific items. Using this Topic as an example, I suggest that both the specific and overall conclusion be stated.

4.27 Topic IX-3, Station Service and Cooling Water Systems

Again, as noted in 4.24, not all the safety objectives given for this topic (see pages A-77 and -78) are addressed. Further, under the heading of Status, reference is made to proposed generic reviews and technical activities. Were these proposals carried out? Additionally, are there any current probabilistic studies of flood hazards and flooding effects which should be noted?

4.29 Topic IX-6, Fire Protection

The application of 10CFR50, Appendix R, concerning fire protection and safe shutdown analysis and compliance, is a major undertaking. All that is noted is that associated circuits will be reviewed generically and outside the context of the SEP.

Additional Comments on Achieving An

"...integrated and balanced..." Decision

From the descriptions given of the methods used to identify topics of safety significance in Phase I of SEP, I conclude that the identification process was thorough, at least for the conditions known up to and including 1977. I appreciate the need for restricting expansion of the topics so that designated goals can be assigned. The developers of SEP recognized that resolution of generic items would proceed independently of SEP, but that somehow there would be an integrationof emerging NRC positions into an effective and efficient molding for the "...final safety assessment of the plant." Since

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the approach to the "final" safety assessment may be asymptotic, I recommend deletion of "final".

Supplements are to be issued to present the status of the generic items. Additionally, the NRC, in special reports to the Congress, prepares the "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants." For example, NUREG-0705, March 1981, identified four new USIs and listed a number of candidate issues for consideration as USIs. (I have not seen the 1982 report.) Along with the moving target of USIs, SEP reviewers need to include the TMI Action Plan Items, a variety of IE Bulletins, and Generic Letters. Attention has to be given to the management and incorporation of the information being generated into overall safety assessments.

The "...integrated and balanced approach..." should recognize the concerns of the Licensees regarding any possible unwarranted diversion of engineering staff from needed tasks. For example, see NUREG-0839, "A Survey by Senior NRC Management to Obtain Viewpoints on the Safety Impact of Regulatory Activities from Representative Utilities Operating and Constructing Nuclear Power Plants," August 1981. On the other hand, a dissenting viewpoint within the NRC, such as given by Demetrious L. Basdekas (published in the Minneapolis Star and Tribune, April 9, 1982) cannot go unchallenged, particularly when he writes that "...the government and industry are unable or unwilling to deal honestly and urgently with far reaching nuclear-safety problems." I know in the past that dissenting views were acknowledged and answered. Perhaps this has already been done for the present case.

Peripheral issues may need to be included. For example, proposed changes for Technical Specifications purport to reduce the number and level of detail in the technical portion and permit the use of a supplementary category. The criteria being developed for these changes should be consulted before implementing resolution of SEP Topics through added Tech Specs.

Only a limited, but useful, application could be made of the probabilistic risk assessment study for this plant. Considering the large NRC budgeted research in this area for the past several years, the emphasis on development of methodologies, and the various current applications, I expect more feedback into the SEP activities.

H. S. Isbin
April 23, 1982

REVIEW OF PALISADES PLANT SEP REVIEW

BY

DR. ZENON ZUDANS

Franklin Research Center
A Division of The Franklin Institute

Z. ZUDANS, PH.D.
Senior Vice President and Chief Operating Officer

April 28, 1982

Mr. W. Russell
SEP Project Manager/Technical Coordinator
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Review of SEP Integrated Plant Safety Assessment Report

References: 1) SECY 76-545
2) SECY 77-561
3) NUREG-0820, Integrated Plant Safety Assessment, Systematic Evaluation Program - Palisades Plant, Consumers Power Company, Docket No. 50-255

Dear Mr. Russell:

In accordance with your request I have reviewed the Palisades SEP program and offer the following comments.

At present, the SEP Program appears to be well organized and well managed. The referenced documents summarizing Palisades SEP review provide a comprehensive historical review of the entire SEP program since its inception in February, 1978 by the NRC.

Significant amount of thought and ffort has been put in development of the procedure for SEP. The procedure a., shown on Figures 1 to 3 was constructed from the Referenced Documents and from various personal conversations with the SEP Program Staff. As it can be seen from Figures 1 to 3, the procedure is generally well defined and at the completion should lead to the satisfaction of the Commission's goals for the SEP program.

In order to make the Integrated Assessment process more responsive to reviewers related human factors, procedure blocks following item (1) (circled) should be provided with more specific guidelines as to the interfacing between SEP, USI, TMI Action Plan, and other personnel involved in the remaining steps of the procedure. This implies that the Final Integrated Assessment Report (FIAR) should be a joint effort of SEP, USI, TMI Action Plan, and others. Details in NUREG-0820 show that such interaction took place for topics of concern to various programs (supplement to NUREG-0820 for resolution of USI, TMI Action Plan, and other items, for example).

SEP review for Palisades, NUREG-0820, is comprehensive and engineering arguments are sound. Following this procedure, Figure 1, 24 topics exit at (1) identified as generic items related to USI and TMI Action Plan, 23 topics exit at (2) because these are not applicable to Palisades and 90 topics reach (3) where the actual review of SEP topics for Palisades begins. It is not identified in NUREG-0820 how many topics were handled by Method 1 and how many by Method 2. However, at the step A (Disposition of Topics), all but 31 were left for backfit candidacy, the remaining 59 having been put in one of the categories 1 to 3. None of the topics fell in the category 4 (i.e., safety significant departure), requiring prompt action. I find that technical arguments leading to distribution of topics to various categories are acceptable.

With respect to Topics in the group of Integrated Assessment, Sections 4.1 to 4.31 of NUREG-0820, represent the Draft Integrated Assessment Report (DIAR). Since Licensee response and resolution for most Integrated Assessment Topics is already contained in NUREG-0820, it also represents Final Integrated Assessment Report (FIAR). However, there is no integration of 24 USI and TMI Action Plan SEP Topics until supplement to NUREG-0820 is issued. This Supplement will also form the basis for conversion from POL to FTOL. In other words, it appears that for Palisades the procedure shown in Figures 1 to 3 was not followed strictly, or stated otherwise, actual Palisades review procedure indicates that plant and licensee specific circumstances may require flexibility in the procedure itself.

With respect to the specific topics reviewed in NUREG-0820, I offer the following additional comments.

For 14 of 31 topics slated for Integrated Assessment, risk assessment by Sandia (SAI) (using Calvert Cliffs unreleased PRA) provided useful insight in relative value of backfits, i.e., it provided logical support for engineering judgment in complicate situations.

Similarly, an extensive use of the plant operating experience in support of engineering judgment is probably the best decision made by the SEP staff. This practice should be followed in the review of other SEP plants.

My overall impression of Palisades SEP Review is that considerably more sound engineering effort has been put in Palisades SEP review, in particular in terms of proper understanding of design, processes and consequences involved, than maybe normally done during regular licensing review process, (SEP Topic list covers essentially all safety related design aspect of a nuclear power plant). The process however, is not complete until all open items are resolved in an integrated manner.

Topic V-11.B is listed in Section 3.1 (final list of 90 topics for Palisades), but it is not referred to in Sections 3.2 (topics meeting current design criteria), 3.3 (meets current criteria because of modifications implemented by the licensee), or Section 4.1 (Integrated Assessment Topics). I believe the substance of Topic V-11.B is addressed in Section 4.16.

I am particularly impressed with the discussion of high and low pressure fluid systems interaction of Sections 4.16 and 4.17. Of the proposed corrective alternatives, Staff's alternative 2 is the best choice in my opinion.

Section 4.7, Topic III-3.C discusses inspection program for water control system structures. I find the five year inspection frequency proposed by the licensee not technically sound (icing season comes once a year).

With respect to Topic III-5.4 (Section 4.9) I am not a strong believer of rigid piping systems. Accordingly, I suggest that the number of pipe whip supports (for postulated 200 pipe break location) should be kept at the minimum.

Section 4.11.1, Topic III-7.A, brings up an important point on need for monitoring forces in individual tendons rather than the average of all tendon forces. Relative to the concrete crack inspection at tendon anchorages, one must note that tendons are always under prestress load which is the bulk of the load ever seen by the tendon. If the tendons are lifted (for tendon force verification), load on anchorages may exceed the load applied due to pressure used for leak rate testing. Accordingly, anchorage concrete inspection should be done at the time when tendons are lifted for force testing (if they are lifted).

Relative to Topic VI-4, I like to point out that the internal pressure load on piping is less significant than the structural loads imposed by geometric constraints of attachment points, I agree with the staff that no backfitting is required.

Topic IX-3, flooding of intake structures, licensee proposes alarms in control room to indicate occurrence of flooding, presumably to give operator time to prevent inundation of service water pumps. If the flooding comes in a form of a 13 ft wave driven by seiche, alarm will not help to prevent water intake structure flooding nor can the operator do much to stop the wave.

For Topic IX-5, the Licensee has proposed (by test or by analysis) to demonstrate the operability of AFW with loss of ventilation in pump room. I doubt that anlaysis can predict realistically the time AFW will remain operable under such conditions. It might be possible to run a relatively short duration test, observe ΔT in room and equipment and make an extrapolation from the test results.

Main steam line break in Containment (Topic XV-2) with postulated single failure is clearly an issue important to risk.

In general, NUREG-0820 provides comprehensive discussion of definition, safety objectives and status of all SEP topics. Adequate list of references is provided for each topic for detail study and for a complete documentation of the decision process.

I am well impressed with the work done by the staff on Palisades SEP review.

Very truly yours,

Zenons Zudans
Senior Vice President

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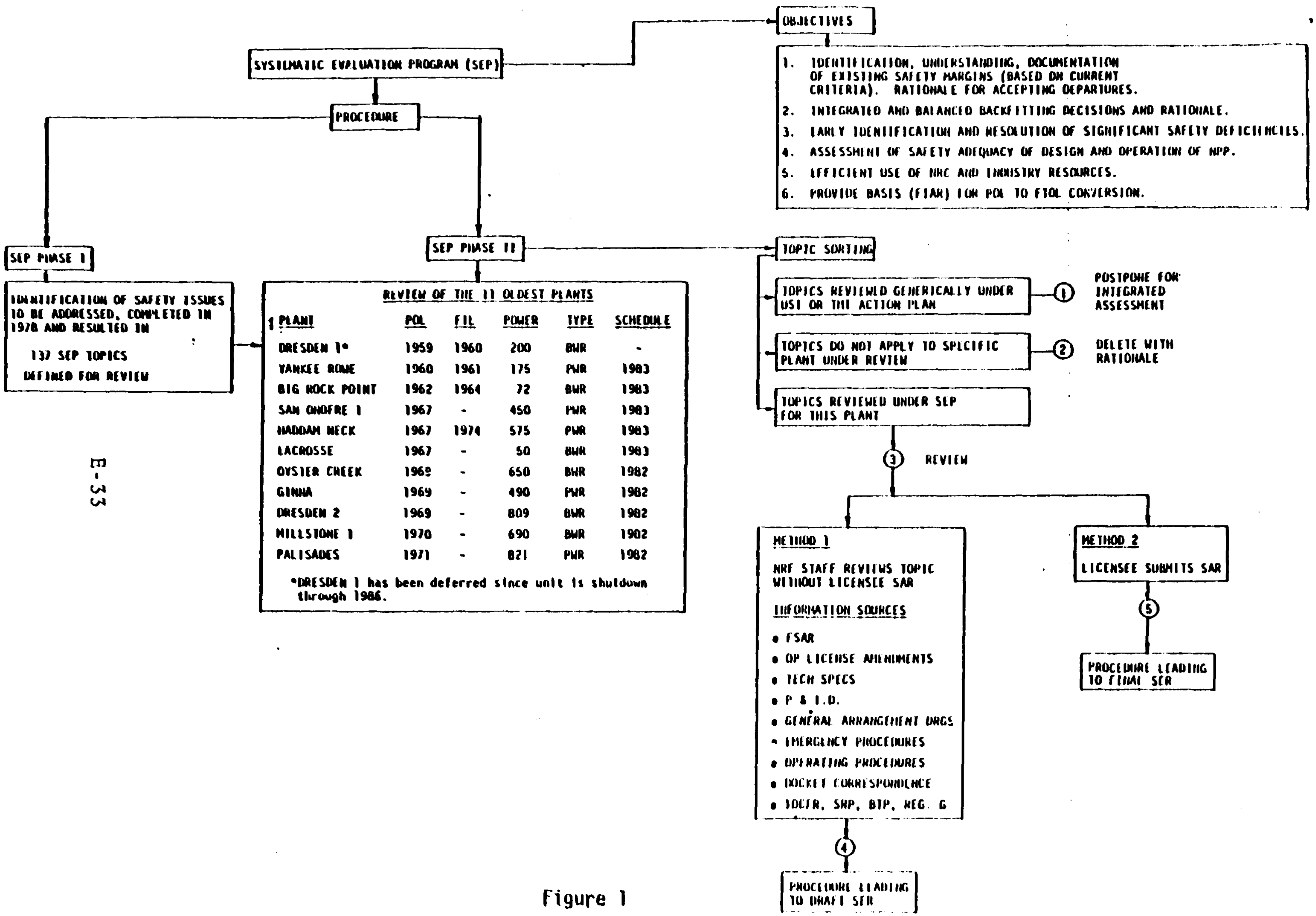
Figs. 1 to 3



PLANT	POL	FTL	POWER	TYPE	SCHEDULE
DRESDEN 1*	1959	1960	200	BWR	-
YANKEE ROWE	1960	1961	175	PWR	1983
BIG ROCK POINT	1962	1964	72	BWR	1983
SAN ONOFRE 1	1967	-	450	PWR	1983
HADDAM NECK	1967	1974	575	PWR	1983
LACROSSE	1967	-	50	BWR	1983
OYSTER CREEK	1969	-	650	BWR	1982
GINNA	1969	-	490	PWR	1982
DRESDEN 2	1969	-	809	BWR	1982
MILLSTONE 1	1970	-	690	BWR	1982
PALISADES	1971	-	821	PWR	1982

Figure 1

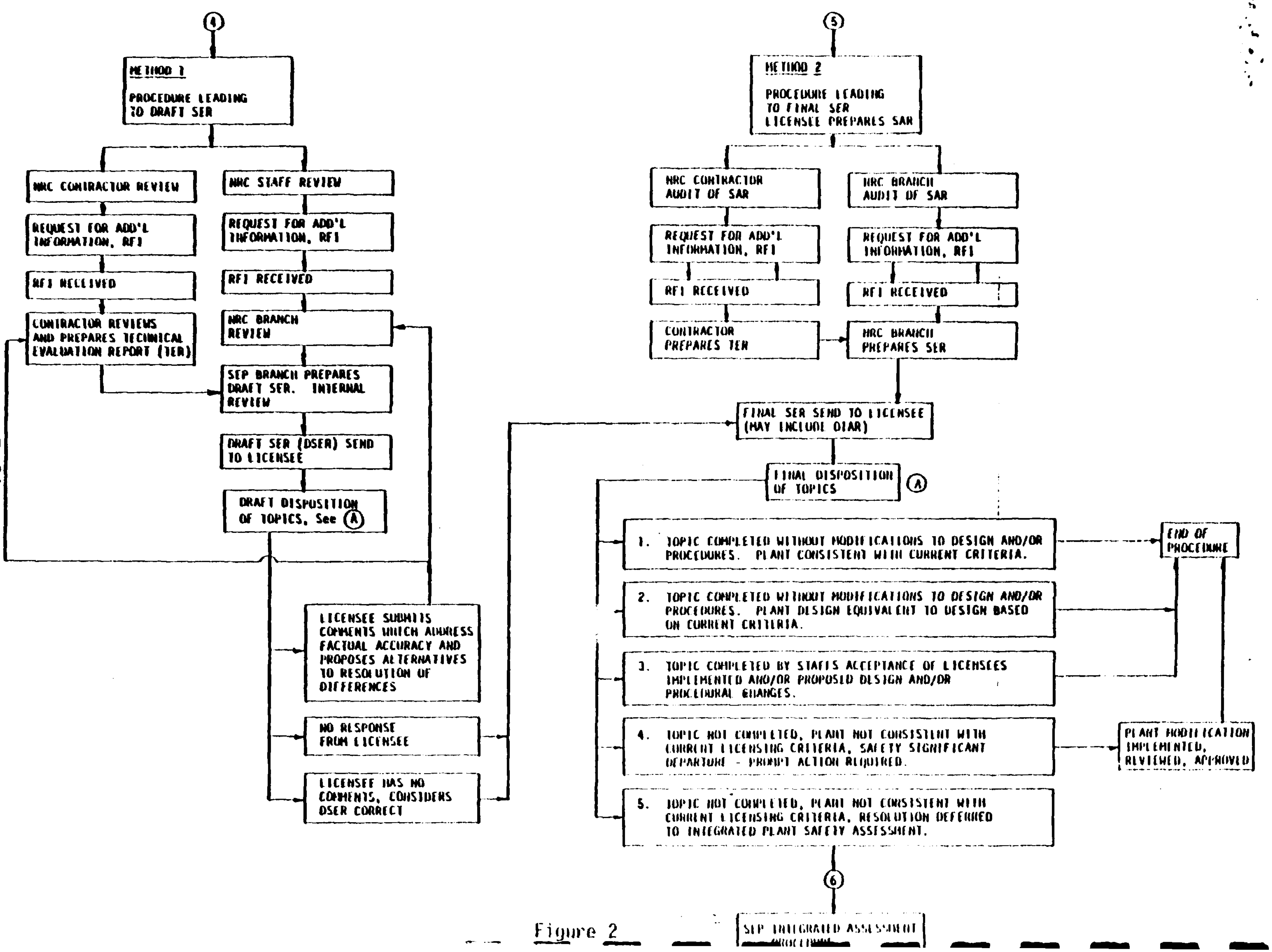
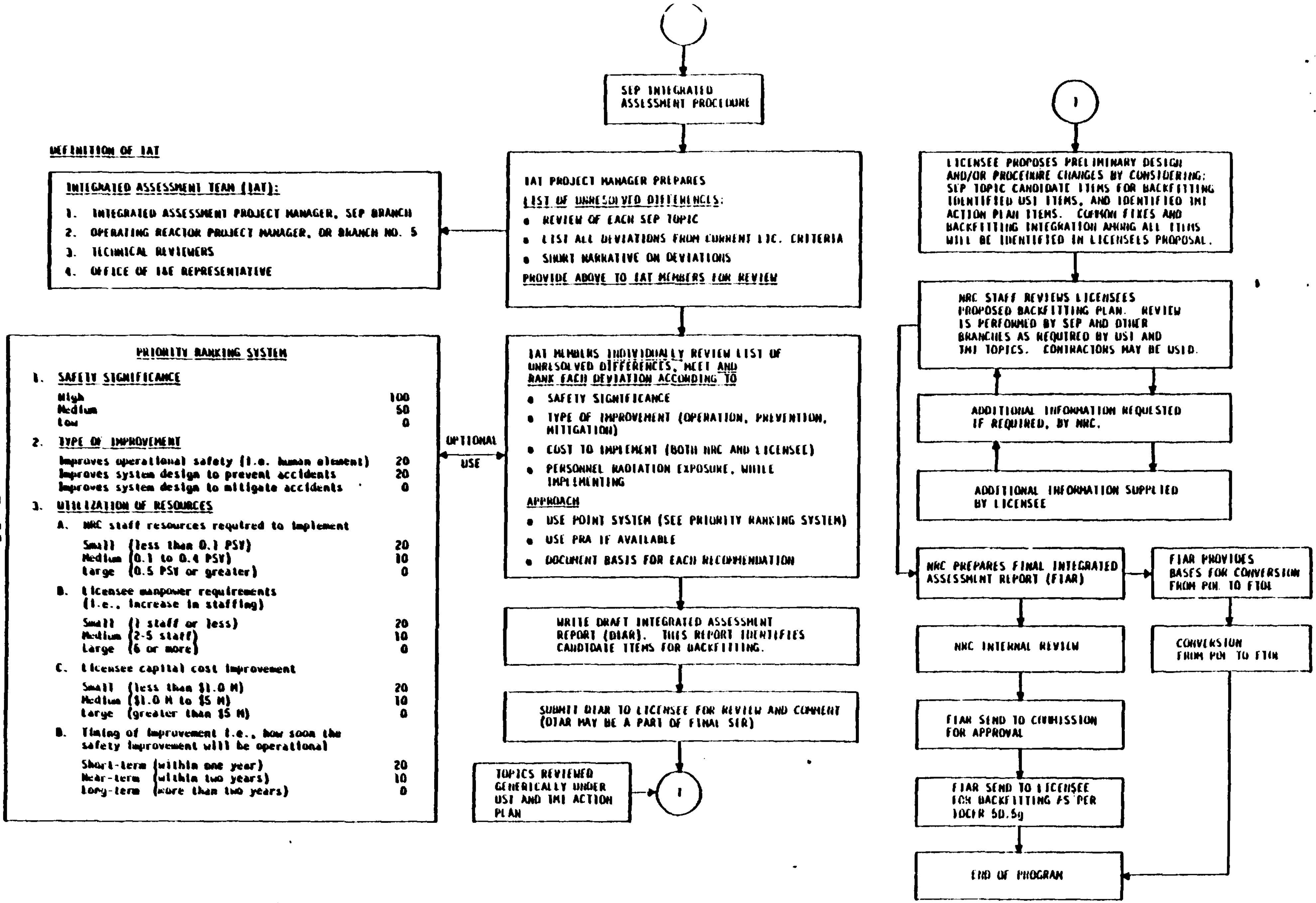

4
METHOD 1
PROCEDURE LEADING TO DRAFT SER
NRC CONTRACTOR REVIEW
REQUEST FOR ADD'L INFORMATION, RFI
RFI RECEIVED
CONTRACTOR REVIEWS AND PREPARES TECHNICAL EVALUATION REPORT (TER)
NRC STAFF REVIEW
REQUEST FOR ADD'L INFORMATION, RFI
RFI RECEIVED
NRC BRANCH REVIEW
SEP BRANCH PREPARES DRAFT SER. INTERNAL REVIEW
DRAFT SER (DSER) SEND TO LICENSEE
DRAFT DISPOSITION OF TOPICS, See A
LICENSEE SUBMITS COMMENTS WHICH ADDRESS FACTUAL ACCURACY AND PROPOSES ALTERNATIVES TO RESOLUTION OF DIFFERENCES
NO RESPONSE FROM LICENSEE
LICENSEE HAS NO COMMENTS, CONSIDERS DSER CORRECT
5
METHOD 2
PROCEDURE LEADING TO FINAL SER LICENSEE PREPARES SAR
NRC CONTRACTOR AUDIT OF SAR
REQUEST FOR ADD'L INFORMATION, RFI
RFI RECEIVED
CONTRACTOR PREPARES TER
NRC BRANCH AUDIT OF SAR
REQUEST FOR ADD'L INFORMATION, RFI
RFI RECEIVED
NRC BRANCH PREPARES SER
FINAL SER SEND TO LICENSEE (MAY INCLUDE DIAR)
FINAL DISPOSITION OF TOPICS A
1. TOPIC COMPLETED WITHOUT MODIFICATIONS TO DESIGN AND/OR PROCEDURES. PLANT CONSISTENT WITH CURRENT CRITERIA.
2. TOPIC COMPLETED WITHOUT MODIFICATIONS TO DESIGN AND/OR PROCEDURES. PLANT DESIGN EQUIVALENT TO DESIGN BASED ON CURRENT CRITERIA.
3. TOPIC COMPLETED BY STAFFS ACCEPTANCE OF LICENSEES IMPLEMENTED AND/OR PROPOSED DESIGN AND/OR PROCEDURAL CHANGES.
4. TOPIC NOT COMPLETED, PLANT NOT CONSISTENT WITH CURRENT LICENSING CRITERIA, SAFETY SIGNIFICANT DEPARTURE - PROMPT ACTION REQUIRED.
5. TOPIC NOT COMPLETED, PLANT NOT CONSISTENT WITH CURRENT LICENSING CRITERIA, RESOLUTION DEFERRED TO INTEGRATED PLANT SAFETY ASSESSMENT.
END OF PROCEDURE
PLANT MODIFICATION IMPLEMENTED, REVIEWED, APPROVED
6
SEP INTEGRATED ASSESSMENT PROCEDURE

Figure 2



Reference: May 15, 1981 letter
D.G. Eisenhut to H.R. Denton

Figure 3

APPENDIX F

MHB/SKI ONGOING SEP ASSESSMENT STUDY

WORK ELEMENTS

WORK ELEMENTS:

1. MHB will continue to follow the plans and progress of SEP and SSEP through meetings and phone contact with the NRC people involved, and obtain reports and letters related to the programs.

2. MHB will attend briefings of the ACRS or Commissioners on the progress of SEP and SSEP. Reports and evaluations will be prepared and sent to SKI.

3. MHB will review and evaluate any items identified by the NRC in this period as being particularly significant to safety and requiring a quick review. These items will be addressed to SKI's attention.

4. MHB will provide monthly progress reports and copies of documents obtained in connection with (1), (2), and (3) plus our assessment of impact of the findings.

5. At the end of 6 months (i.e., approximately June, 1981), the progress will be assessed and the study evaluated in terms of direction and content.